UNIVERSITY OF UTAH
NUCLEAR REACTOR FACILITY
LICENSE NO. R-126
DOCKET NO. 50-407

SAFETY ANALYSIS REPORT, TECHNICAL SPECIFICATIONS, AND ENVIRONMENTAL REPORT

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUR OR DENOTED BY BRACKETS

THE UNIVERSITY OF UTAH

June 01, 2009

Dongok Choe
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50 S. Central Campus Drive Rm 1206
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Document Control Desk
U.S. Nuclear Regulatory Commission

RE: License Renewal and Power Up rate of the University of Utah Nuclear Reactor Facility License number R-126, Docket number 50-407

Dear,
We reviewed the previously submitted material (SAR-Safety Analysis Report) for University of Utah TRIGA reactor renewal and we decided that the SAR should not be withheld from public disclosure.
This SAR should not be withheld from public disclosure.
We are transmitting the updated complete SAR for University of Utah TRIGA reactor.
Please discard the SAR which provided in the July 2006.

If you have any further questions, I can be reached at the University of Utah through the following contacts:

Dongok Choe, Ph.D.
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Best regards,

Dongok Choe
Reactor Administrator/Reactor Supervisor
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THE UNIVERSITY OF UTAH

July 6, 2006

Melinda Krahenbuhl
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Document Control Desk
U.S. Nuclear Regulatory Commission

RE: License Renewal and Power Up rate of the University of Utah Nuclear Reactor Facility
License number R-126, Docket number 50-407

Please replace in entirety Ch 4, 13 and appendices A and C, from the original submission dated March 25, 2005.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on July 6, 2006,.

Respectfully,

Melinda Krahenbuhl PhD
Director /Reactor Administrator
University of Utah

Cc: M. Mendonca

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THE UNIVERSITY OF UTAH

March 25, 2005

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Document Control Desk
U.S. Nuclear Regulatory Commission

RE: License Renewal and Power Up rate of the University of Utah Nuclear Reactor Facility
License number R-126, Docket number 50-407

We respectfully request renewal of the class 104 facility operating license for the University of Utah nuclear reactor, revised to permit operation for a further twenty years at a nominal maximum steady state power level of 250 kW. The Reactor Safety Committee has reviewed all documents supporting the request. The applicant, University of Utah, is a non-profit educational institution and an agency of the State of Utah. There is no foreign control of the University of Utah. To support this request we are transmitting the following documents.

- Safety Analysis Report
- Financial Qualifications
- Technical Specifications
- Environmental Report
- Emergency plan
- Requalification Program

Correspondence regarding this application should be addressed to M. Krahenbuhl (address above).

Respectfully,

Melinda Krahenbuhl

Melinda Krahenbuhl PhD
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UNIVERSITY OF UTAH

TRGA

SAR

Chapter 1

The UUTR Facility

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List of Figures and Tables

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1 THE UUTR FACILITY

1.1 Introduction

The purpose of this Safety Analysis Report (SAR) is to support an application to the Nuclear Regulatory Commission (NRC) from the University of Utah (U of U) for renewal of its TRIGA Modified Mark I Nuclear Reactor, License R-126, Docket 50-407 and request an increase of the maximum power to a thermal power of 250 kW at steady state operation without pulsing. The entire TRIGA reactor facility is contained within the "Center for Excellence in Nuclear Technology, Engineering, and Research" (acronym - CENTER), and the TRIGA reactor is denoted herein as the University of Utah TRIGA Reactor (acronym - UUTR.) The reactor facility is located on the main campus of the University of Utah in Salt Lake City, Utah. The proposed license renewal and upgrade will benefit all interested parties served by the facility within the regional area. The UUTR reactor core contains standard TRIGA fuel (enrichment <20%), is cooled by natural convection, and exhibits a large negative temperature coefficient or reactivity that provides inherent safety characteristic of all TRIGA reactors. The UUTR reactor is license, administered, and operated by the University of Utah for education, research, and reactor services. This SAR documents the University of Utah's intent to continue operation and maintenance of the UUTR reactor under NRC regulation.

This document, supported by previous documents filed with the NRC including earlier SAR's, Technical Specifications, Safety and Security Plans, and Environmental reports, provides the description of the reactor system and its associated components' details, the general features, characteristics, and basic operation of the reactor. These documents reflect the as-built condition and the current administration and operation history of the facility. These documents include experience with the operation and performance of the reactor systems, radiation surveys, and personnel exposure histories related to operation of the UUTR at 100 kW. These analyses utilize conservative assumptions to provide ample operating safety margins. The descriptions and analyses contained in these reports is deemed to provide sufficient information to assure that the health and safety of the public will be protected with continued operation of the reactor as proposed in this SAR.

1.2 Summary and Conclusions on Principle Safety Considerations

The physical operating safety of the UUTR reactor is based upon the TRIGA reactor fuel characteristics, the existing instrumentation, and the control and safety systems as presently implemented for the UUTR. In addition, this document will show that abnormal operating conditions and potential accidents for the UUTR reactor are similar to other TRIGA reactors. Therefore, the UUTR poses no unreviewed issue or risk to the health and safety of the public.

The UUTR reactor fuel, control-rod drives, control rods, and experimental systems are equivalent to other TRIGA reactors sited throughout the United States and the World. These components and systems have well established operating experience, and no additional reactor design review is required for the proposed license renewal and power upgrade.

The UUTR facility incorporates numerous safety characteristics that are designed to insure safe and conservative reactor operation and performance. The UUTR reactor is housed in a secure, structural building especially designed for supporting and maintaining engineering laboratories. Additionally, the building is located over two hundred feet above the surrounding valley flood plain. [redacted]. The bottom of the aluminum tank is a 1.0 cm welded aluminum plate. The aluminum material used for the bottom of the inner tank is 6061-T6 aluminum. Construction and welds on the inner aluminum tank comply with ASME unfired pressure vessel standards, and both the inner and outer tanks are water tight. The bottom of the aluminum tank rests on a reinforced 60 cm concrete pad that forms the bottom seal for the outer steel tank which is 1020 mild steel coated and sealed to minimize corrosion. The roof over the facility is the second floor of a three story engineering laboratory building. This floor is a 10 cm poured concrete slab over a 0.7 cm structural steel ribbed support liner. The concrete floor is load bearing and designed to support heavy engineering machinery.

The building housing the UUTR was designed to conform with Uniform Building Code Zone 3 criteria and the UUTR containment tanks and supporting structures also comply with Uniform Building Code Zone 3 criteria. [redacted]

[redacted] The UUTR reactor can withstand large positive reactivity insertions and loss-of-coolant accidents without a significant release of fission products. The low radiation exposures documented in earlier UUTR SAR's and other TRIGA operating experience associated with the design-basis accidents, demonstrate that the UUTR and supporting structure and building are adequate for continued safe operation of UUTR.

Fire detection and suppression systems are installed throughout the engineering building and reactor area. In addition, the UUTR facility is equipped with [redacted] radiation monitoring systems. Radiation-monitoring equipment has been installed at key locations to monitor radiation levels and to sound alarms and scram the reactor if preset values are exceeded.

The reactor room equipment crane is designed and constructed in accordance with OSHA 29 CFR Part 1910.184, Overhead and Monorail Cranes. All crane components have been designed for resultant static loads based on rated capacity with a minimum factor of safety of five based on the ultimate strength of the crane components. The fuel transfer cask lifting lugs are designed using the ANSI/ASME code as guidelines. Design analysis shows a margin of safety greater than six when the entire load of the cask is held by one of the two lifting lugs. In addition, all of the fuel transfer equipment is load tested, maintained and operated in accordance with ANSI/ASME during all fuel handling operations. This design, fabrication and testing approach, coupled with the low exposures associated with fuel element clad failures, shows that this system is adequate for its intended use. The UUTR reactor water shield is similar to other TRIGA reactors but exhibits a minimum height of water above the core top of 6.5 m (~22 feet). This level of water provides attenuation for 1 MeV gammas from the core to the water surface by a factor of exp(-36)=E-16 where the mass attenuation coefficient for water is 0.06 (sq cm/g). Furthermore, this level of water provides attenuation for fast neutrons from the core to the water surface by a factor of exp(-62)=E-27 where the fast neutron removal cross section for water is 0.103 (1/cm).

The reactor water shielding and supporting containment provide biological shielding adequate to maintain personnel exposures ALARA and protect the reactor from natural and human disasters. The reactor room air handling system maintains the reactor room at a negative pressure with respect to surrounding areas to control and prevent the spread of airborne radioactive materials. The air from the reactor room passes through HEPA filters prior to being discharged to the atmosphere. In the event of a release of radioactive material within the reactor room, the reactor room air handling system automatically closes the inlet dampers effectively isolating the room by increase the negative pressure and preventing the release of activity to surrounding offices. Sufficient air is discharged through the HEPA filters and CAM system to maintain negative air pressure in the reactor room.

The inherent safety of the reactor lies primarily in the large, prompt negative temperature coefficient of reactivity characteristic of the TRIGA fuel-moderator material. Thus, even when large sudden insertions of positive reactivity are made and the reactor power rises on a short period, the prompt negative reactivity feedback produced by the increase in fuel temperature causes the power excursion to be terminated before the fuel approaches its safety limit temperature and the fuel cladding is breeched. The prompt shutdown and safety characteristics of reactors fueled with TRIGA fuel have been demonstrated during transient tests conducted at General Atomic, Inc. in La Jolla, California as well as other pulsing facilities. This demonstrated safety has permitted the siting of TRIGA fueled reactors in urban areas in buildings without the pressure-type containment usually required for power reactors.

Abnormal conditions or postulated accidents that are addressed in this SAR and earlier documents include the Maximum Hypothetical Accident (MHA), reactivity insertion, loss of coolant, loss of coolant flow, fuel cladding failure. For reactivity insertion, loss of coolant, and the loss of coolant flow accident scenarios (using actual measured reactivity worths), fuel and cladding temperatures remain at levels below those required to produce cladding failure; thus, no release of fission products will occur.

The limiting fault condition, which assumes failure of fuel cladding and an air release of fission products from one fuel element, will result in radiation doses, both thyroid and whole body, to operations and base personnel and to the general public that are well below those allowed by ANS 15.7.

Operating experience which documented radiation exposures to personnel working in the UUTR from both direct and airborne radiation during normal operation have been reviewed and assessed. These analyses and measurements show that the exposure rates for the proposed power increase to 250kW are well within NRC accepted exposure limits. Under normal operating conditions, personnel will be subjected to a maximum radiation field of less than 1 mR/hr. In actual practice, radiation exposures will be lower since typical operation times are much less than the conservative assumptions indicate. All personnel entering the facility will be closely monitored, exposures kept to as low as reasonable achievable (ALARA), and in no case will exposures be allowed to exceed 10 CFR Part 20 regulations or guidelines.

The effects of a single fuel element clad failure during normal operation have been evaluated for both operating personnel and the general public. The results show all exposures below the 10 CFR Part 20 limits.

The radiation exposures from Ar-41 (half life ~ 1.8 hr) and N-16 (half-life ~ 7 s) produced during normal operation of the reactor have been evaluated for both operating personnel and the general public. These short-lived isotopes result in maximum exposures of only a few mR/yr. to maximally exposed operating personnel within the immediate area of the reactor. Their release to the atmosphere, through the UUTR stack, results in a maximum down wind concentrations well below the 10 CFR Part 20 guidelines for unrestricted areas.

1.3 General Facility Description

The reactor is located in the Center for Excellence in Nuclear Technology, Engineering and Research (CENTER) facility on the ground floor of the Merrill Engineering Building on the University of Utah campus, within the city limits of Salt Lake City. The reactor has been operated safely for ~30 years at the current licensed maximum thermal power level of 100 kW. It is proposed that the reactor can be operated at variable thermal power levels up to a maximum licensed power of 250 kW at steady-state with no pulsing operations.

1.3.1 Location of Facility

The UUTR is located in the Merrill Engineering Building (MEB) on the north end of the University of Utah Campus in Salt Lake City, Utah. The 1,167 acre campus is situated east of the city center on the foothills of the Wasatch Mountains. Figure 1.3.1 shows the campus in relation to Salt Lake City and other towns and communities situated in the Great Salt Lake Valley.

1.3.2 Characteristics of the Site

The University of Utah TRIGA Reactor is located in [redacted] the Merrill Engineering Building (MEB), which is a three level structure located on the campus of the University of Utah. [redacted] The MEB is situated on high ground and rises about 100 feet above the nearest neighboring structures. The building is constructed with structural steel frames and reinforced concrete floors acting as diaphragms in distributing loads to vertically resisting elements. The MEB conforms to seismic zone 3 requirements of the Uniform Building Code. MEB was designed by Dean L. Gustavson Associates, Architects, and constructed by Alder Child Construction Company.

The building has approximately 254,778 square feet of floor space assigned as follows: Classrooms - 24,859; Offices - 25,547; Teaching laboratories - 59,399; Research laboratories - 97,847; workshops, storerooms, corridors, etc. - 56,126. The areas immediately above the reactor room, on the second floor, are faculty and departmental offices. Radiation surveillance of the areas immediately above the UUTR has been performed by the Radiation Safety Officer at the University of Utah since the initial licensing of the UUTR. These areas will continue to be monitored. Residence by any person in this area can be controlled if necessary. [redacted] The third floor above the reactor area is comprised of office and laboratory space.

The reactor area comprises eight rooms including the reactor control room, computational laboratory, radiation measurement laboratory, radiochemistry laboratory, microscope room, reactor chemistry laboratories, radioactive storage, and reactor room (see Figure 1.3.2 A.) [redacted] The reactor room and the laboratories will be treated as a single unit for ventilation and safety confinement purposes. The walls in the area are constructed of one-hour fire-resistant, standard, plaster and metal stud construction with the exception of the west wall, which is an exterior reinforced concrete wall. Large windows provide visibility to the reactor room from the administrative offices, control room and computational lab. Nonporous enamel paint finishes were used on all walls and ceilings of the reactor area for ease of clean up.

The UUTR reactor is cooled by natural-convection of the treated light water coolant.

The core includes graphite and heavy water reflectors and is designed to accommodate the source ends of three beam tubes. [redacted] Direct visual and mechanical access to the core and mechanical components are available from the top of the tank for inspection, maintenance, and fuel handling. The water provides adequate shielding for personnel standing at the top of the tank. The three control rod drives are mounted above the tank on a bridge structure extending from the east side of the tank.

The UUTR contains the electrical, water, and sewer utilities required for operation. In addition, the facility has fire detection and suppression systems, intercom communications, radiation monitoring systems, security systems, irradiation facilities, and fuel handling equipment.

1.3.3 Design Criteria, Operating Characteristics, and Safety Systems

The general design of the UUTR follows the philosophy of defense in depth. The facility is contained in an engineering laboratory building that meets to seismic zone 3 requirements under the Uniform Building Code. [redacted] The core design and layout is such that cooling by natural convection meets the fuel's temperature limits, established at 1000 °C for stainless steel clad fuel. The fuel also exhibits a large negative temperature coefficient so that in the event of a prompt excursion the reactor will inherently and passively shut down the reactor without fuel damage. The fuel is in a zirconium hydride matrix that has demonstrated high fission product retention with both stainless steel and aluminum cladding.

The UUTR safety systems include multiple scram capabilities for elevated temperature or power, elevated radiation levels, loss of electrical power, and low water level. Control rod interlocks exist to limit the reactivity insertion rates. In addition to the scram systems, the ventilation system maintains a negative pressure in the facility during operation. This ventilation system also operates in an emergency mode to mitigate a radionuclide release.

1.3.4 Engineered Safety Features

The general design of the tank and shield are shown in Figure 1.3.2B. For this design, the neutron flux is shielded by 75 cm (2.5 feet) of water in the tank and 60 cm (2 feet) of sand surrounding the inner tank to minimize induced radioactivity in local soil outside the secondary tank. Shielding composed of 22 feet of water, concrete, sand and/or earth surround the core providing adequate radiation protection for operating personnel at any location in the reactor room.

The reactor tank consists of an inner aluminum tank and an outer steel tank. The failure of either tank from severe seismic disruption or human event (aircraft crash) would still provide sufficient residual water within the surviving tank to provide adequate radiation shielding for personnel within the immediate area. All surfaces of the steel and aluminum tanks that come in contact with the soil have been coated to inhibit corrosion. The bottom of the tank is well above the local ground water level, which is several hundred feet below the reactor tank floor.

Since the bottom of the aluminum tank rests on a reinforced concrete pad that forms the bottom seal for the outer steel liner, the maximum stress that the bottom of the aluminum tank is subject to is the hoop stress at the bottom weld. Under a hydrostatic head of 24 feet of water, the hoop stress at the bottom of the inner aluminum tank is 24.1 MPa (3500 psia). The hoop stress on the outer steel tank, if subject to the same head, is 68.9 MPa (l0,000 psia). These figures are well below the yield stress of 170 MPa for aluminum and 390 MPa for steel (Appendix A.1 Tank Stress).

The facility exhaust systems are designed to maintain the reactor room at a slightly negative pressure with respect to surrounding areas and the exterior of the building. This action prevents the uncontrolled release of radioactive contamination to adjacent areas in the engineering building. These systems can maintain concentrations of radioactive gases in the reactor room to levels that are well below the 10 CFR Part 20 limits for restricted areas.

Fuel for the UUTR reactor is comprised of the standard TRIGA reactor fuel elements that are enriched to less than 20% U-235. TRIGA reactor fuel is characterized by inherent safety, high fission product retention, and the demonstrated ability to withstand water quenching with no adverse reaction in temperatures of up to 1150°C. The inherent safety of TRIGA reactors has been demonstrated by extensive experience acquired from similar TRIGA systems throughout the world. This safety arises from the large prompt negative temperature coefficient that is characteristic of uranium-zirconium hydride fuel-moderator elements used in TRIGA systems. As the fuel temperature rapidly increases, this negative reactivity immediately compensates for positive reactivity insertions. This results in an inherent and passive mechanism whereby reactor power excursions are terminated quickly and safely.

1.3.5 Instrumentation and Control

The UUTR reactor console employs digital electronics for signal processing and display. The console is located in the reactor control room and supports all reactor operations including control rod movement operations, temperature and radiation monitoring, safety settings, interlocks, and scram functions. The console displays information on control rod positions, power level, fuel temperatures, reactor period, and other system parameters. The console also performs many other functions, such as monitoring reactor usage and water quality display. In addition to the above instrumentation, a video camera is located above the tank to provide full surveillance of the core and components. The monitor for this camera is located on the control console. In addition, electrical power to the console and reactor safety and radiation monitoring equipment has a battery back system to insure safe operation in the event of loss of building electrical power.

1.3.6 Reactor Cooling System

The reactor tank has a capacity of approximately 8000 gallons with a water depth from the surface to the top of the core of approximately 6.5 m (22 feet). The reactor core is cooled by the natural convection flow of light water within the reactor pool. Water purity is maintained by a dual bed demineralizing system including particulate filters and resin beds. Makeup water is added to the reactor tank from the main building water supply as necessary by passing the makeup water through the demineralizing system.

A shell and tube heat exchanger rated at seven-and-a-half ton capacity can be used to cool the pool water. The heat exchanger receives warm water from the pool and returns cooled water. Both the inlet and return lines to the pool have a small l/4 inch hole approximately one foot below the normal pool water level. These holes prevent siphoning of the pool water below that level siphon inlet. The water pumped to the heat exchanger is circulated at the rate of about 20 gallons per minute by a centrifugal pump. The reactor pool temperature can be maintained at about 16° C (60° F), which is approximately the ambient ground temperature. For any operation at power above 25 kW, the heat exchanger-cooler has insufficient capacity to maintain the 16 °C (60° F.) When operating at 100 kW, the pool temperature rises approximately 3° C per hour (5° F per hour) and at 250 kW the pool temperature will rise at 8° C per hour (14° F per hour). Administrative control will limit the maximum water temperature to 40° C (104° F). After the reactor is shutdown, the cooling system can lower the pool water temperature approximately 0.6° C per hour (1° F per hour.)

1.3.7 Radioactive Waste Management and Radiation Protection

The University of Utah has a Radiological Safety program on campus. The University's Radiation Safety Officer is located here and provides radiological services and monitoring for the entire University including the Medical School, researchers employing radioactive materials in their work, and the UUTR. The University's Broad Form Radiation License issued by the State of Utah (an NRC Agreement State) is administered by Radiological Health Department. All radioactive wastes generated within the UUTR facility are disposed of through RHD as specified in 10 CFR 61, and 71, and following the provisions as outlined under NRC agreement state status. In addition, RSD maintains oversight on all personnel dosimetry, and radionuclide transport as required in 10 CFR 20, and 49 CFR 171 to 178.

1.3.8 Experimental Facilities

1.3.8.1 Central Irradiation Facility

The UUTR central irradiation facility is located in the central fuel pin position of the TRIGA core. A special tube has been constructed to accommodate samples and can be placed in the central fuel pin position by means of a cable. The dimensions of this assembly are the same as a fuel pin.

1.3.8.2 Beam Ports

The reactor system permits the insertion of up to three diagonally directed beam tubes that extend from the reactor core to the reactor floor. There are no plans to open the diagonal beam ports. The upper terminus of each beam tube is adequately shielded with sand and secured with a 1/8 inch thick steel locked cap at the reactor floor level. The beam ports are presently closed and filled with sand and have not been installed or used to date.

1.3.8.3 Pneumatic Transfer Facility

A pneumatic transfer system is available for rapid irradiations at the UUTR facility. The rabbit is installed within a 1.5 inch O.D. tube and is driven by dry, compressed nitrogen. The pneumatic rabbit tube has a curved trajectory to prevent direct streaming of radiation from the core to the surface of the pool.

1.3.8.4 Dry Tubes

A dry tube thermal irradiator is available for use in a trapezoidal shaped D_2O tank attached to the side of the reactor core. The samples are placed into polyethylene vials attached to fishing line and dropped into the irradiator through a curved PVC tube that extends to the top of the reactor pool.

1.4 Shared Facilities and Equipment

The UUTR is located in the Merrill Engineering Building on the University of Utah campus. Within this building, water and electrical power is provided from a centralized source. The water purification for the UUTR, is operated and maintained by the UUTR staff. University of Utah maintenance staff maintains the dedicated ventilation system for the facility. In the event of a loss of power, a battery back up system can provide emergency power to the reactor system for at least 24 hours.

1.5 Similar Facilities

The design of the UUTR fuel is similar to other TRIGA type reactors. The functional characteristics of the UUTR Reactors' Instrumentation and Control System are also similar. Thirty-nine of these reactors were constructed in the late 1950's and 1960's. Since a large number of these reactors have been in operation for many years, considerable operational experience is available and their characteristics are well known and documented. As of February 2003, 19 TRIGA's are operational in the US. There are 23 TRIGA reactors in operation world-wide and 2 presently under construction.

The UUTR control rod drives, with the exception of the motor driver, are essentially the same as control rod drives used on other TRIGA systems. The UUTR drives use a stepping-type electric motor rather than the non-synchronous, single-phase motors used on earlier drives. The UUTR design and operation for the stepping-type motor has been fully developed and tested and is used on the Sandia National Laboratory TRIGA Reactor System.

1.6 Summary of Operations

The UUTR provides a wide range of training, irradiation, and research services to education, research, and industry. The reactor has experimental facilities for the irradiation of materials. These facilities support the irradiation services that include isotope production for medical and industrial clients, neutron activation analysis, neutron damage testing of electronic components, ultra-sensitive detection of actinides, and educational support. Currently, the beam ports are sealed and not in use. To support the regional services, the UUTR typically operates about four hours per week at 90 kW. The operational work load is not expected to increase significantly from this level. However, increasing the power level to 250 kW expands the variety of experiments that can be provided and reduces run time for certain standard experiments (e.g., fission track analysis and neutron activation analysis).

1.7 Nuclear Waste Policy Act of 1982

In accordance with the Nuclear Waste Policy Act of 1982, the UUTR Licensee has signed a contract with the DOE for the return of the reactor fuel that is owned by the DOE to the DOE at cessation of the UUTR License.

1.8 Facility Modifications and History

The University of Utah has operated a reactor on campus for approximately 37 years. The current reactor has been operating successfully since 1975 when it was licensed to operate at 100 kW with no pulsing capabilities. The UUTR at the CENTER in the University of Utah is the only operating research reactor in the State of Utah.

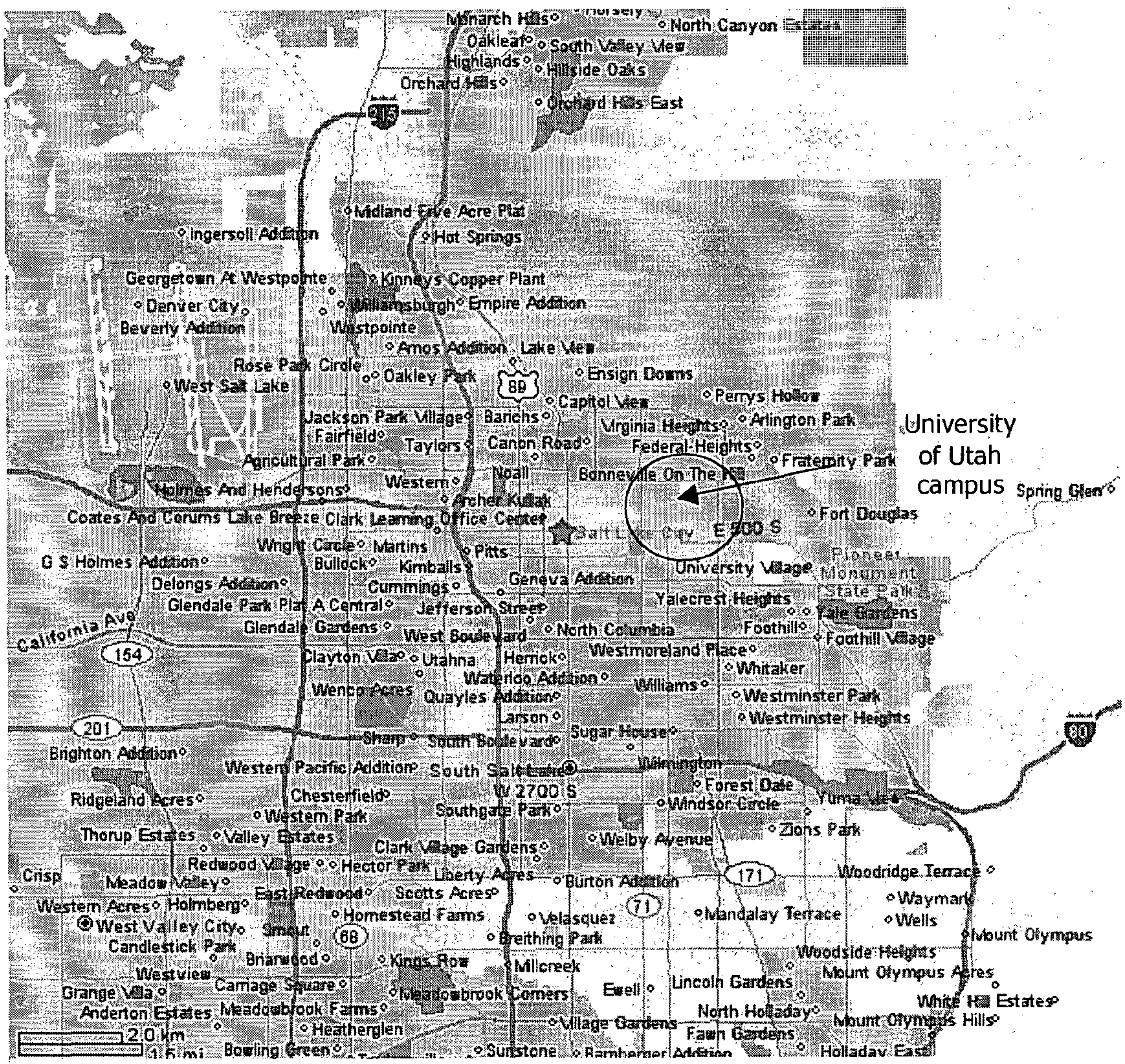

Figure 1.3.1

University of Utah and Surrounding Area

[redacted]

[redacted]

Chapter 2

Site Characteristics

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(Figures and Tables are numbered according to the sections that refer to them.)

2 SITE CHARACTERISTICS

2.1 Introduction

This chapter provides information on the site characteristics of the University of Utah and vicinity as they relate to the safety considerations of the UUTR reactor. The conclusion reached in this chapter and throughout this document is that the selected site is well suited for the UUTR facility when considering the inherently safe operating characteristics of the reactor including the Maximum Hypothetical Accident (MHA). This is consistent with the conclusions reached for the other TRIGA reactors operating throughout the world, many of which are located on university campuses, in hospitals and other highly populated areas.

2.2 Geography and Demography

2.2.1 Site Location and Description

The reactor is located in [redacted] the Merrill Engineering Building (MEB) on the north end of the University of Utah Campus in Salt Lake City, Utah. The 1,167 acre campus is situated east of the city center on the foothills of the Wasatch Mountains. Figure 1.3.1, in Ch. 1, shows the campus in relation to Salt Lake City and other towns and communities situated in the Great Salt Lake Valley.

The UUTR reactor has several boundary associated with its location. Moving from the inside to out, the most interior region is the reactor room. [redacted]. Normal access to this area is through the reactor control room. Personal dosimetry is required in this area. [redacted]. The next boundary is the CENTER laboratory. This laboratory is a controlled area supported by a daytime staff.

2.2.2 Population Distribution

Salt Lake City is the largest city in the state of Utah with an official population of 181,743 (2000 census). Salt Lake County, which has an area of about 737.4 square miles, has a population of about 898,387. The area surrounding the MEB is lightly populated with some residences about 300 meters west and 600 meters north of the building. No permanent dwellings lie east of the building since the campus extends in this direction up the slopes of the Wasatch Mountains. South of the MEB, the campus extends for about one mile with former federal government land now owned by the University extending another half mile south. The nearest residences to the south are approximately 1.5 miles from the building. No student dormitories on the campus are closer than 2,000 feet to the reactor site. According to the U of U Radiological Health Deaprtment, the daytime campus population of faculty, students, and staff is estimated at 25,000 to 30,000 people. The nighttime population is estimated at 5,000 people. According to the U of U Office of Residential Living, the number of students living on campus in residence halls is 1,250. The Merrill Engineering Building has a total daytime occupancy of approximately 500, while the nighttime occupancy is approximately 30 people.

2.3 Nearby Industrial, Transportation, and Military Facilities

The UUTR is located on the northern side of the campus. North, Northwest, and Northeast: located 400 meters from the facility are private residences; these houses are situated on quarter acre lots and extend north approximately one mile before terminating in the mountains. West and Southwest: located 100 meters from the facility are private residences; these houses are situated on quarter acre lots and extend north, west, and, south approximately two miles before merging with the built up area of downtown Salt Lake City; beyond downtown, lies the Salt Lake International Airport with an Air National Guard Transport Wing approximately 10 miles west. East and Southeast: the University of Utah Medical Complex is east; this Complex has three hospitals and numerous medical facilities; the medical complex is located more than 500 meters distance from UUTR, and this complex terminates to the east in the mountains; the medical complex extends south and merges into university dormitories (1000 meters) and the U. S. Army reserve post, Fort Douglas (1200 meters); beyond Fort Douglas lies University of Utah Research Park which is a complex of research laboratories and one hospital (2000 meters). South: the University campus extends south for approximately 1000 meters with dormitories on the far south and south-southeast (800 meters); beyond the campus to the south is a hospital and then residential housing.

Only one road is located adjacent to the facility, North Campus Drive. This road moves traffic in an east west direction. It turns north where it passes closest to the facility (50 meters), and then continues east west. North Campus Drive is a four-lane road (two lanes each direction) with a 35 mph speed limit. Typical traffic consists of hospital and university employees, and students.

2.3.1 Air Traffic

Both the University of Utah Hospital and the Primary Children's Medical Center 1 km east of the UUTR have heliports. The Salt Lake International airport does not have standard flight paths over the reactor site or in the vicinity of the university because of the high student population on campus.

2.3.2 Effects of Potential Transportation or Hospital Accidents on the Facility

Hospital accidents and transportation on North Campus Drive cannot affect the reactor. Accidents occurring at Fort Douglas and the Research Park also cannot affect the reactor. Additionally, neither of these complexes uses North Campus Drive for industrial transportation. The airports are situated such that private flight routes over the facility are extremely rare, and the hospitals heliports are small and used only for shock trauma patients. This limited accident potential and the submerged design of the reactor greatly reduce the risk of any credible external accident threatening the UUTR fuel.

2.4 Meteorology

Aside from altitude (approximately 4,200 feet mean sea level) and the Wasatch Mountains, the most influential natural condition affecting the climate of Salt Lake City is the Great Salt Lake. This large inland body of water, which never freezes due to its high salt content, tends to moderate the temperature of cold winter winds blowing from the west and northwest. Of lesser influence are the Oquirrh Mountains located twenty miles to the southwest. This range, with several peaks above 10,000 feet, shelters the Salt Lake Valley somewhat from storms associated with southwesterly winds. Due to the Wasatch Mountains, about three to five inches more precipitation per year can be expected along the eastern edge of the city, including where the University of Utah is located, than over the valley a few miles to the west.

Salt Lake City has a semi-arid continental climate with four well-defined seasons. Summers are characterized by hot, dry weather. The high temperatures during this season are usually not oppressive, since the relative humidity is generally low and the nights usually cool. July is the hottest month with average maximum temperatures in the nineties. Table 2.4 contains the recorded weather normals, means, and extremes of temperature and precipitation for the Salt Lake City. Weather data is from Utah Meteorology Department.

2.4.1 Temperatures

The average daily temperature range is about thirty degrees in the summer and eighteen degrees during the winter. Temperatures above 102° in the summer or colder than 10° below zero in the winter are likely to occur one year out of four.

2.4.2 Precipitation

Winters are cold, but usually not severe. Mountains to the north and east act as a barrier to frequent invasions of cold continental air. The average annual snowfall ranges from 52 inches at the airport to over 70 inches in the foothill area of the eastern portion of the city. Similarly, the average maximum depth of snow during the winter varies from 9 to about 13 inches. The average duration of continuous snow cover is 29 days.

Precipitation, generally light during the summer and early fall, reaches a maximum in spring when storms from the Pacific Ocean are moving through the area more frequently than at any other season of the year.

2.4.3 Humidity

The Salt Lake Valley is a semi arid climate where summer humidity seldom exceeds 20%.

2.4.4 Winds and Stability

Winds are usually light, although occasional high winds have occurred in every month of the year, particularly in March. Figure 2.4.4 A indicates the prevalent wind directions for Salt Lake City. When the air circulation is predominantly of local origin, the air moves from the Great Salt Lake towards the mountains during the sunlight hours and complements the daytime up slope valley winds. During the night, the down slope mountain winds flow toward the Great Salt Lake from all the surrounding mountains. This diurnal wind pattern is the result of mountainous terrain and the Great Salt Lake. During the daytime, the wind near the University campus is from the west to southwest; during the nighttime, the wind is from the north to northeast. Super adiabatic conditions exist during the afternoon with strong inversion conditions being established soon after sunset. At sunset, the wind speed is low and nearly constant with height as the change from up slope wind to down slope wind takes place. The temperature profile changes rapidly from super adiabatic in the afternoon to an inversion soon after sunset. The profiles for the morning show a reverse condition when the down slope winds change over to upslope winds as a result of the heating of the mountain slopes. The morning change in wind direction is less abrupt. It takes from one to two hours after sunrise to heat the mountain slope enough to destroy the strong surface inversion. Winter storminess destroys local temperature differences and the highly disturbed motions of storms tend to dissipate non-frontal inversions. Conversely, as a storm moves away and a high-pressure cell develops over or moves into the local area, strong inversions tend to form sometimes quite rapidly and persistently. The summers are characterized by weak and stagnant pressure systems usually with a high-pressure center fairly near Salt Lake City or a flat pressure system over Utah. These high pressures with weak, large-scale gradients and clear or nearly clear skies create favorable synoptic situations for inversions. Figure 2.4.4 B shows the percentage of the time the inversion height is less than 7000 ft in the Salt City area.

However, the release of radioactivity during a temperature inversion does not pose a threat to the populous regions surrounding the U of U to the west since any contaminant retained in the inversion would be removed by the predominant midmorning upslope winds that move up and along the spur mountain range to the north of the campus and by the upslope winds that flow into Red Butte canyon to the east-southeast. In general, any release of airborne radioactivity would be rapidly dispersed and flow into the surrounding canyons away from the population center because of the prevailing wind flow patterns during working hours (0800 to 1700). Figure 2.4.4 C shows up slope and down slope wind flow patterns on the Wasatch Front.

2.4.5 Severe Weather

In the spring, Salt Lake City often experiences high winds. These winds can sometimes exceed 80 mph but they are only found in the far southern portion of the valley. UUTR is located on the northern end of the valley with high mountains in close proximity to the north and the east. The location of the mountains and northern location of the reactor cause the winds to be much less severe at the reactor site. Winds seldom exceed 40 mph on site and the facility has witnessed 20 years of windstorms without incident.

2.5 Hydrology

Hazards from activation of ground water are considered remote. Ground water was not encountered in any of the borings made at the building site (see section 2.6.1). Furthermore, the concrete and water shielding around the core minimize neutron activation of the surrounding soil.

2.5.1 Floods

The Merrill Engineering Building which houses the reactor facility is located on the campus on high ground sloping westward with an average slope of about ten percent. Drainage characteristics around the building reactor site are favorable, and no threat of flooding or water damage exists.

2.6 Geology, Seismology, and Geotechnical Engineering

Salt Lake City lies in the northeastern part of the Jordan River Valley, which is surrounded on three sides by mountains and opens toward the northwest into the Great Salt Lake basin. The Oquirrh Range makes up the western boundary of the valley with an average height of 8,500 ft. above mean sea level and peaks to 9,700 ft. mean sea level. The southern boundary consists of the low east-west traverse range split by a water gap containing the Jordan River. The eastern boundary of the valley is formed by the Wasatch Range with higher summits near 12,000 ft. mean sea level. A spur of mountain extends a short distance westward from the main Wasatch Range just north of Salt Lake City. Two main canyons break the Wasatch Range in this area. In addition, many smaller canyons open into the valley from all ranges. The valley floor consists of two parts: a broad, flat central plain at an elevation of approximately 4,225 feet mean sea level and a system of narrow Lacustrine terraces that intervene between the central flat and the bordering mountains. The plain and the surrounding terraces are part of the former lake bottom of prehistoric (late Pleistocene) Lake Bonneville.

Utah has a high earthquake frequency and contains several active fault zones. The Coast and Geodetic Survey (ESSA) of the U. S. government has classified the seismicity region through Central Utah, which includes Salt Lake County, as zone 3. Figure 2.6 shows the major fault lines within the state of Utah. The University of Utah Seismograph Station provided this figure.

2.6.1 Site Geology

Information on the foundation soil is available from the "Foundation Investigation for the Merrill Engineering Building", U/U Job No- SA-261, submitted to the Utah State Building Board in November 1957. Figure 2.6.1A shows the general site geology. The results of 3 of the borings made at the site are given in Figure 2.6.1B. Boring No. 1 was made near the vicinity of the proposed reactor tank excavation. Soils at the site are predominantly granular soils with silty sand (fine) to clay silt binder. Coarser particles range from sand sizes to gravel, cobbles and boulders. Excavation of the fuel storage pits confirmed the bore data. Standard penetration tests were performed consecutively at intervals of approximately 5 feet in depth. The tests were performed by driving a standard 1 3/8" ID split-barrel penetration sampler one foot into the undisturbed soil by means of a 140 lb. drop weight falling 30 inches. Penetration resistance is logged in terms of blows per foot of penetration. This data is recorded to the right of each log boring in Figure 2.6.1 B. Key to the boring log data is given in Figure 2.6.1 C, and Figure 2.6.1 D gives the safe bearing capacity for footing design at the site. Calculations indicate that the load bearing capacity of foundation soil at the reactor site is more than adequate to support the reactor tank under operation conditions. Actual bearing stress at the reactor tank footing is about 1.9 lbs/ft^2 compared to a bearing capacity of well over 5 lbs/ft^2. No free ground water was encountered at any boring at or in the vicinity of the Merrill Engineering Building.

2.6.2 Seismicity and Maximum Earthquake Potential

In light of the precautions taken during the design and construction of the reactor and selection of the reactor site, there is not a substantial risk associated with the effects of seismic activity on the TRIGA reactor. The emergency most likely to be caused by a severe earthquake is a possible breach or rupture of one of the pool tanks resulting in a reduction of water that shields the reactor core. The simultaneous failure of both tanks is not considered to be a credible outcome. A risk assessment of these scenarios was performed, and the results are reported in Chapter 13 of this document.

Since 1964, The Utah Seismograph Station has recorded 86 earthquakes originating within a 5 mile (8.5 km) radius of the reactor. The largest Richter magnitude recorded for these 86 earthquakes is 2.7, with 98% less than 2. The historical record 1800 - 1964 indicate a further 44 earthquakes with a magnitude 7 earthquake occurring in the early 1900's. Data for the magnitude 7 earthquake is incomplete. Expanding the radius to 20 miles the number of earthquakes for the same time period is 1018. The largest magnitude recorded in this area is 5.2 with 95% below 2. The historical record for the 20-mile radius indicates that 58 earthquakes occurred including the 7 previously mentioned.

Only two damaging earthquakes have occurred in Salt Lake County during the past 146 years. A series of shocks with peak magnitude of 5.5 struck the city on 22 May 1910, and an earthquake of magnitude 4.3 occurred on 4 February 1955. Damage was minor in both instances. The most severe earthquake recorded in the state occurred on 14 November 1901 in Richfield, Utah (about 150 miles south of Salt Lake) and consisted of about 35 shocks of peak magnitude 6.7. The most costly earthquake (magnitude 3.8) occurred on 30 August 1962 in Logan, Utah (about 80 miles north of Salt Lake.) Property damage resulting from this earthquake was estimated at about $1,000,000 and consisted of broken windows, cracked plaster, toppled chimneys, etc. There is no record of any deaths directly related to an earthquake in the State of Utah.

2.6.3 Surface Faulting

The local seismic conditions of the reactor site are generally favorable. The nearest fault is the splinter East Bench Fault, which lies about 2,000 feet (609.6 m) west of the reactor site. Furthermore, in the last 146 years only 6 earthquakes have had their origin at the East Bench Fault, and these have been mild and of local effect only. None had a Richter magnitude greater than 4.9. Thus, as specified by ANSI 15.7 the UUTR is farther than 400 meters from the surface location of a known capable fault.

2.6.4 Liquefaction Potential

To some extent, the type of ground on which a structure is built and the nature of the structure has a greater impact on the earthquake safety factors than proximity to a fault zone. In the case of the UUTR, the reactor site is a dry, compact mantle of earth covering bedrock. According to a 1994 study of liquefaction potential in the Salt Lake Valley conducted by the Utah Geological Survey, the probability of soil liquefaction occurring at the reactor site is very low (less than 5% given that an earthquake of sufficient magnitude to cause soil liquefaction will occur within the next 100 years.) The reactor system is composed of dual containment tanks, which will move with the earth as a unit. The outer steel tank and its heavy reinforced concrete pad can withstand tensile, compressive, and shear forces of short duration. The inner aluminum tank, which contains the reactor core and shielding water, is sufficiently ductile to respond to earthquake motion without tank rupture. The construction of the reactor tank, concrete pad, footings, and structure conforms to zone 3 requirements of the Uniform Building Code.

Table 2.4
Area Weather Norms, Means and Extremes

	Temperature means (°F)			Temperature Extremes (°F)		Precipitation inches						
						Rain				Snow		
	Max	Min	Aver	High	Low	Mean	Max	Min	Max (24 h)	Mean	Max	Max (24 h)
Jan	37	19	28	62	-22	1.3	2.9	0.1	1.4	13	30	10
Feb	43	24	34	69	-14	1.2	2.8	0.1	.9	10	28	11
Mar	52	31	41	78	2	1.8	4.0	0.1	.9	11	42	12
Apr	62	38	50	85	15	2.0	4.6	0.4	1.6	6	26	12
May	72	46	59	93	25	1.7	4.8	0.1	1.3	1	8	5
June	83	54	69	104	35	0.9	2.8	T	1.8	T	T	T
July	93	62	78	107	40	0.8	2.6	T	2.3	0	0	0
Aug	90	61	76	104	37	0.9	3.7	T	1.6	0	0	0
Sept	80	51	65	100	27	1.1	7.0	T	2.3	T	4	4
Oct	66	40	53	89	16	1.3	3.9	0	1.3	2	20	14
Nov	50	30	40	75	-14	1.3	2.6	T	1.1	6	27	10
Dec	38	22	30	67	-15	1.4	4.4	0.1	1.2	13	35	13
Annual	64	40	52	107	-22	15.6	24.3	8.7	2.3	63	100	14

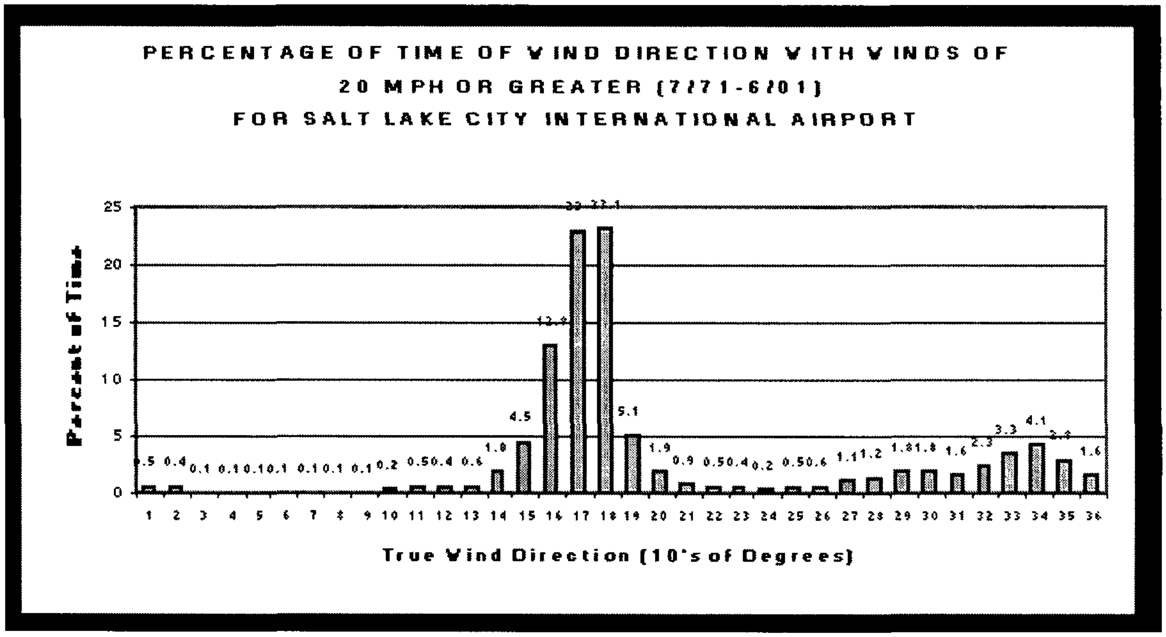


Figure 2.4.4 A

Prevalent Wind Direction for the Salt Lake Area

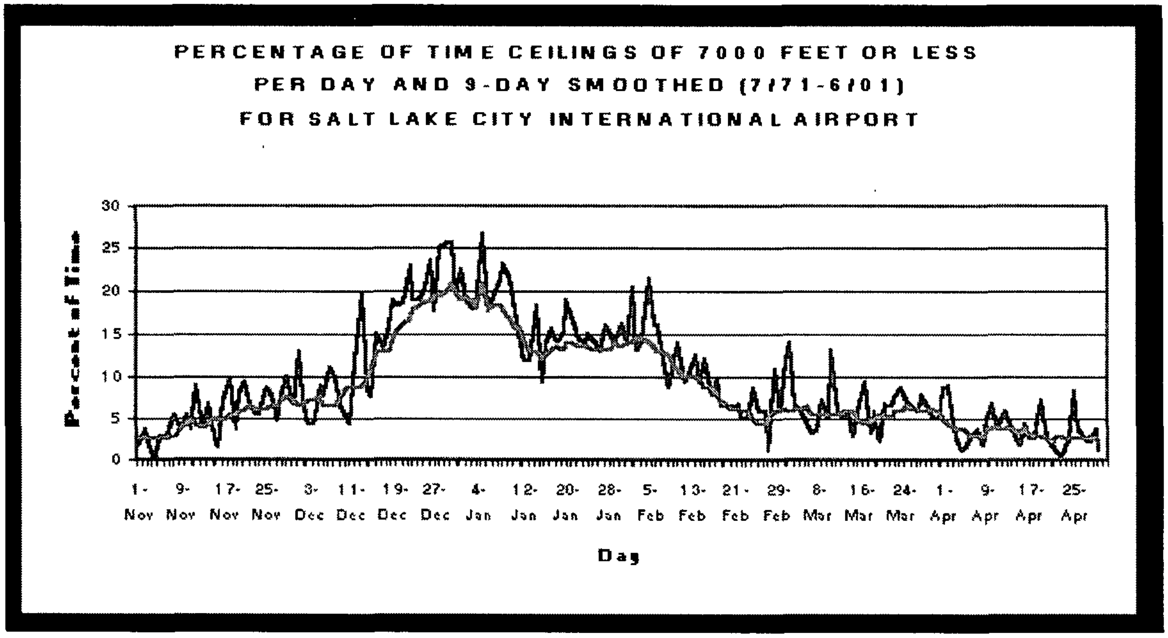


Figure 2.3.2.4 B

Percentage of time that the inversion height is less than 7000 ft.

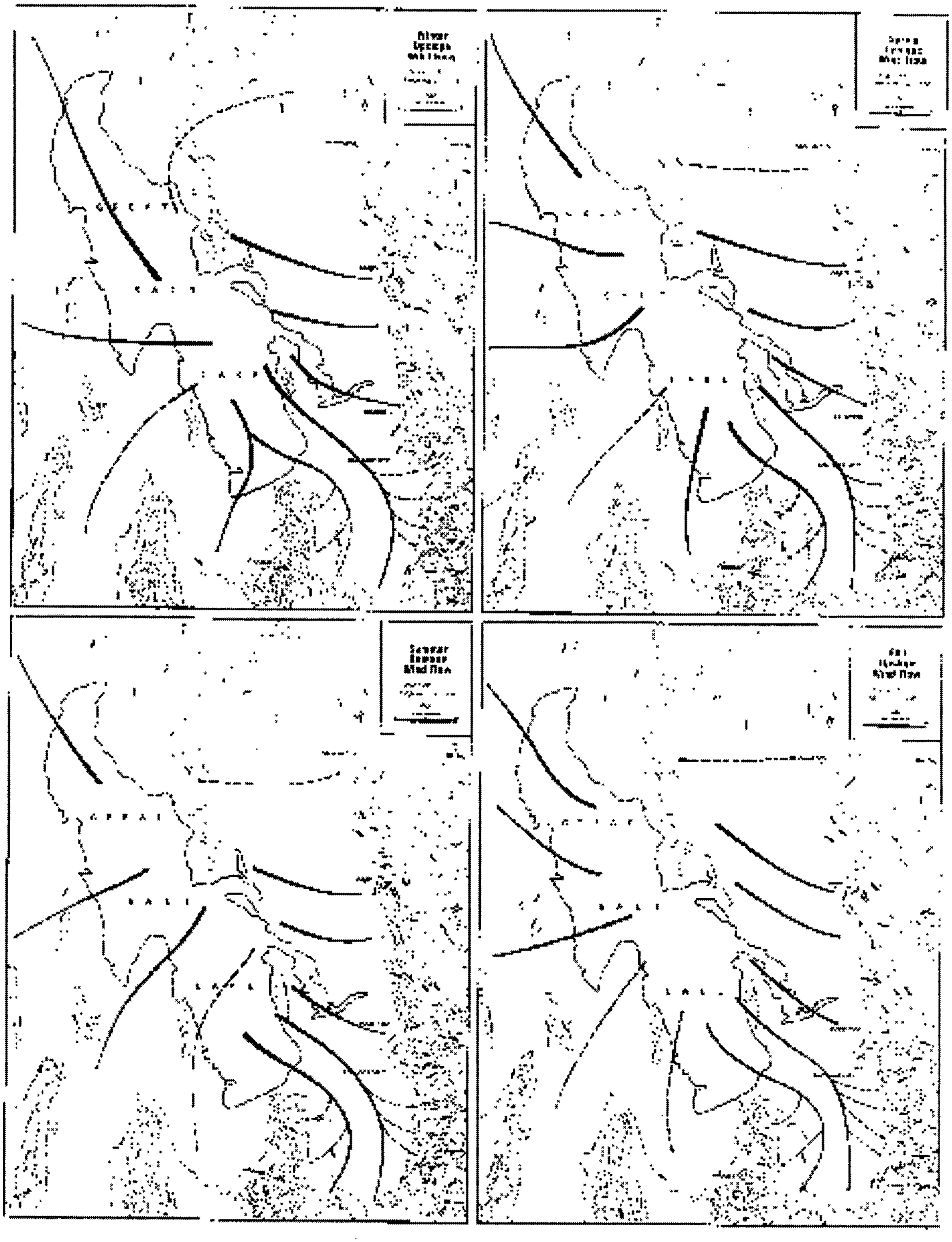


Figure 2.4.4 C

Diurnal Wind Pattern

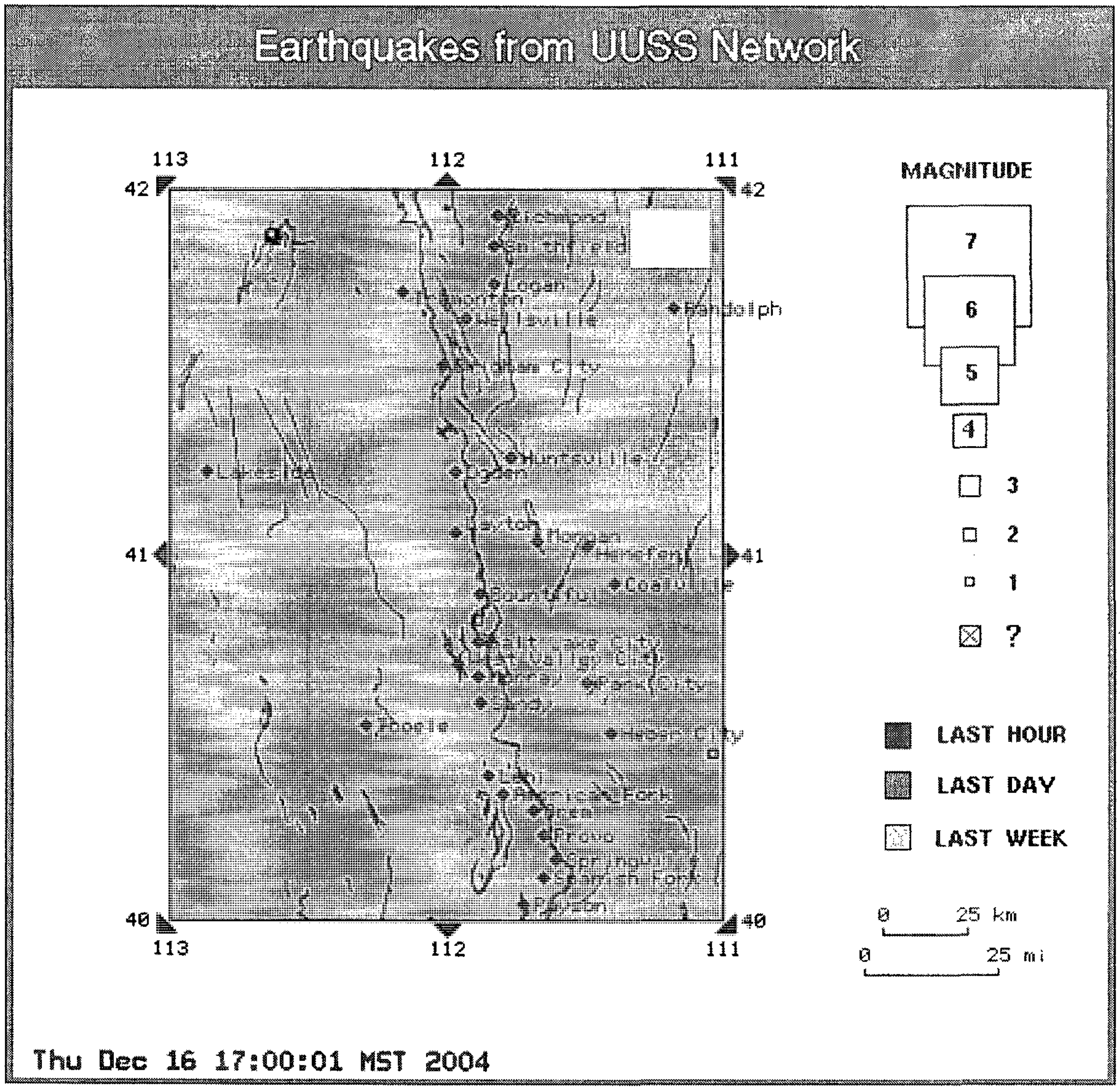


Figure 2.5.1

Fault Lines on the Wasatch Front

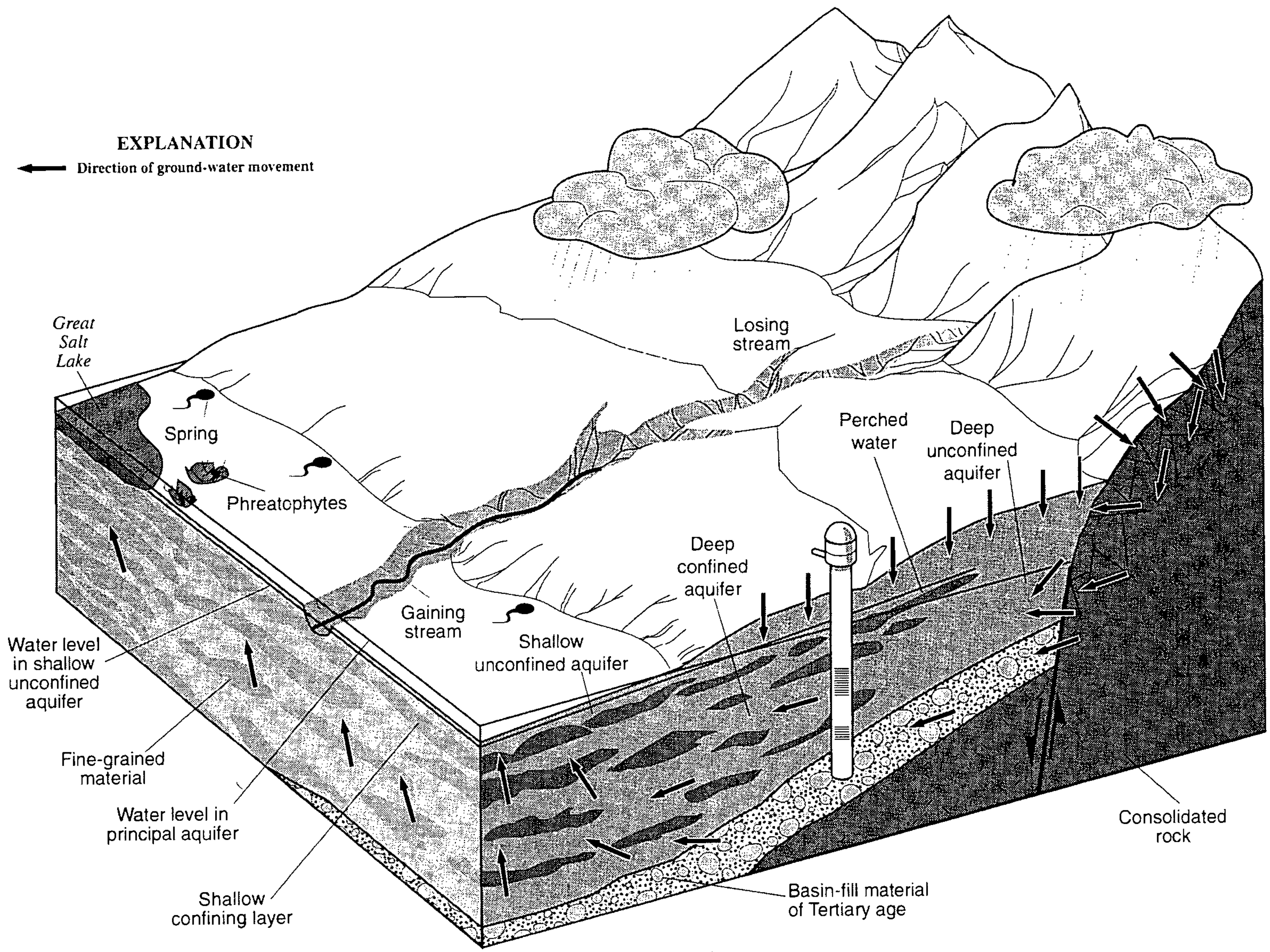
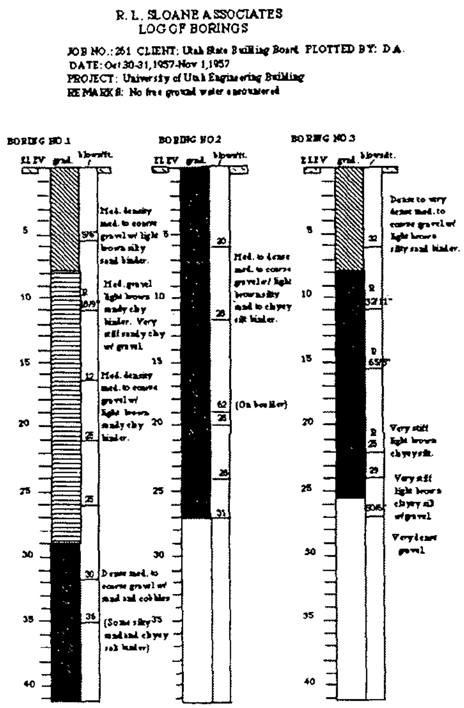
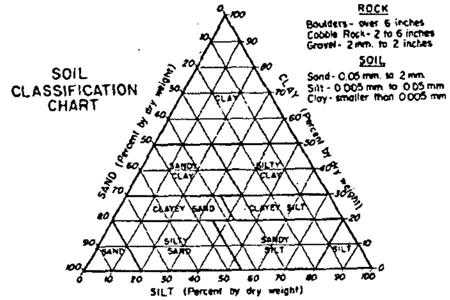


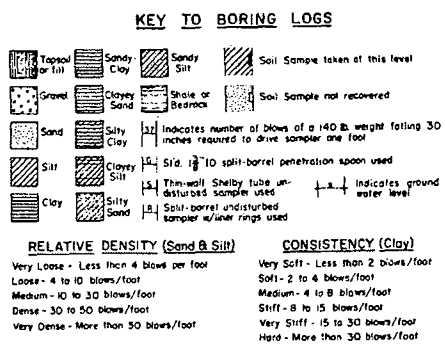
Figure 2.61 A
UUTR Site Geology

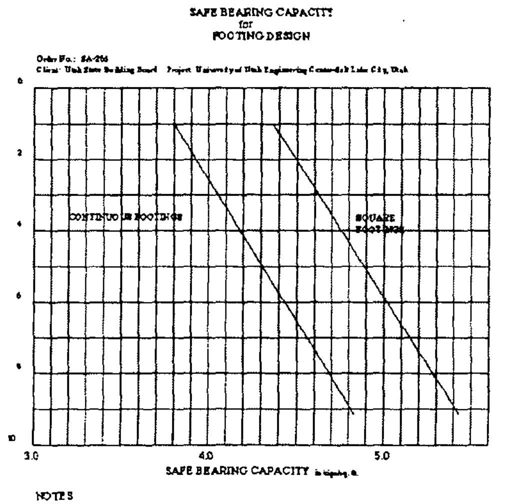
Figures 2.6.1B; 2.6.1C; 2.6.1 D

Log of Borings; Key to Log of Borings; Safe Bearing Capacity for Footing Design









Chapter 3

Design Of Structures, Systems, And Components

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3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Introduction

This chapter discusses the "General Design Criteria for Nuclear Power Plants" as set forth in 10 CFR 50, Appendix A, as they apply to the UUTR. These General Design Criteria were formulated for the purpose of establishing minimum requirements for principal design criteria of water-cooled nuclear power plants. Furthermore, they are to be applied to construction of new plants similar in design for which construction permits have been previously issued. Since the UUTR is a research and testing reactor, many of the power plant criteria cannot be logically applied. Therefore, the discussions of this chapter are oriented towards a separate consideration of each individual criterion, rather than toward identification of areas of noncompliance and corrective actions.

The nominal UUTR steady-state power level is currently 100 kW up-rating to 250 kW. Thus, the fission-product inventory is orders of magnitude less than those of conventional power reactors for which the General Design Criteria were primarily prepared. A conservative upper limit of energy released for an entire year of operation would be about 1 MW-day. These comparisons illustrate why the UUTR may be placed in a category of much lower risk and treated accordingly in a rigorous review of compliance with the General Design Criteria.

The accidents described in Chapter 13, Accident Analysis, conservatively demonstrate that instrumented shutdown actions and building confinement are not necessary to ensure that radiological doses will not exceed allowable limits. Table 3.1.1 presents a synopsis of the conclusions regarding the relevance of the General Design Criteria to the UUTR.

3.1.1 Overall Requirements (Criteria 1-5)

Criterion 1: Quality Standards and Records

Original structures, systems, and components important to safety were designed, fabricated, constructed, and/or tested to design specifications (MAN-NDI-86-03) and associated standards.

University of Utah engineers monitored the construction to assure that the specifications the construction work were in accordance with these specifications. Modifications have been made in accordance with existing standards and requirements.

Criterion 2: Design Bases for Protection Against Natural Phenomena

Hurricanes, tsunamis, and seiches do not occur in the Salt Lake Valley area. Flooding in the valley area could be caused by run-off from local rainstorm activity or snowmelt from the Wasatch Mountains. However, the UUTR is situated some 400 ft above the 100-yr. flood plain, and no local depressions are capable of holding floodwater.

Only a small number of tornadoes, one or two per year, have been reported in Utah. Based on the small probability of occurrences, postulated low intensity, the intermittent type of reactor operation and low fission-product inventory, no criteria for tornadoes have been established for the UUTR structure. However, the buildings are designed to withstand the area wind loads.

The Salt Lake Valley area is classified as being in Seismic Zone 3 as defined in the Uniform Building Code. The UUTR structures have been designed and constructed in accordance with this code, see Chapter 13. Seismic activity in the region has registered as high as 7 in historical time, which indicates an upper limit on the most likely seismic event (refer to Section 2.5). Since the UUTR is designed to the Uniform Building Code for Zone 3, there is ample conservatism in the design for the maximum expected event. The UUTR structures may suffer some damage from a seismic event of the highest possible yield, but, as previously noted, low frequency of operation and low fission-product inventory minimize the risk of such an event, and resultant radiological doses would be within the ranges evaluated in Chapter 13, Accident Analysis.

Criterion 3: Fire Protection

The reactor room and reactor control room structures, built of steel, concrete, and concrete block, are highly fire resistant. However, material inventories inside the rooms could include various flammable materials (paper, wood, etc.); and these, coupled with potential ignition sources, require that fires be considered.

Several features reduce both the likelihood and the consequences of a fire.

1. Periodic fire safety inspections are made by the Fire Marshall.
2. Periodic in-house inspections are made for the explicit purpose of reducing nonessential combustible material inventory.
3. Fire detection and suppression systems are installed in the facility.
4. A control room window into the reactor room permits the reactor operator to continuously observe the reactor room, so that immediate action can be taken to minimize the effects of a fire; established emergency procedures would be put into effect in the event of a fire.
5. The large volume of water in the reactor tank would protect the core from any conceivable fire.
6. The reactor is fail-safe and will shutdown should fire damage the instrumentation or control system.

[redacted]

Criterion 4: Environmental and Missile Design Bases

The construction of the facility precludes catastrophic rupturing of the reactor tank. There is no source within the reactor room for generating large, sustained, positive pressure differentials that could cause a breach of the reactor tank walls.

The amount of combustible materials allowed in the reactor facility has been limited to preclude damage to the reactor should the materials ignite. The closed beam tubes are angled upwards away from the core. This tubing does not penetrate the inner tank wall. There are no high-pressure systems within the reactor room. The piping systems are anchored and do not penetrate the tank walls, and they could not conceivably affect the reactor.

Criterion 5: Sharing of Structures, Systems, and Components

Electrical power and make up water constitute the only systems shared by the UUTR with other users. Sharing is based on the fact that the UUTR electric power is supplied by the power source for the Merrill Engineering Building (MEB). Loss of power results in the shutdown of the reactor since all control circuits are fail-safe, and no power is required for safe shutdown or to maintain safe shutdown conditions. An electric power failure at any point in the UUTR network will not detrimentally affect the reactor. [redacted] Make up water is also supplied from the MEB's water system. This is not a significant safety issue because of the slow seepage rate from the reactor in the case of a tank wall rupture. (See Chapter 13.)

3.1.2 Protection by Multiple Fission - Product Barriers (Criteria 10-19)

Criterion l0: Reactor Design

Accident analyses presented in Chapter 13 show that under credible accident conditions, the safety limit on the temperature of the reactor fuel would not be exceeded. Consequently, there would be no fission product release that would exceed 10 CFR Part 20 allowable radiation levels.

Criterion 11: Reactor Inherent Protection

Because of the fuel material (U-ZrH) and core design, there is a significant prompt negative temperature reactivity coefficient. Routine steady-state power operation is performed with the safety, shim, and regulating rods partially withdrawn. As shown in Chapters 4 and 13.

Criterion 12: Suppression of Reactor Power Oscillations

Not applicable. Due to the small dimensions of the core and low power levels, the reactor is inherently stable to space-time and xenon oscillations.

Criterion 13: Instrumentation and Control

The instrumentation and control system for the UUTR is a modified General Atomics TRIGA Reactor console. The console provides a safety channel (percent power with scram), a redundant, multi-range, linear power safety channel (source level to full power with scram), a wide-range log percent power channel (below source level to full power), a source count rate channel (fission chamber with control rod interlocks), period indication, two fuel temperature channels (C and D rings with scram), and two pool water temperature channels (bulk pool water and refrigeration water). Additional scrams are triggered by a loss of pool water level, a high voltage interrupt tripping the external security system, and airborne radiation levels in excess of the high area radiation monitor's limits. The UUTR can be operated only in manual mode. Percent withdrawal indicators show control rod position and movement. Interlocks prevent the movement of the rods in the up direction under any one the following conditions: scrams not reset, source level below minimum count, two UP switches depressed at the same time.

Criterion 14: Reactor Coolant Pressure Boundary

The reactor tank and cooling systems operate at low pressure and temperature. The vessel is open to the atmosphere, and there are no means for pressurizing the system. [redacted] The primary coolant system components are aluminum, stainless steel, or PVC. The system components outside the reactor tank have a low probability of serious leakage or of gross failure that could propagate. Furthermore, the design of the system is such that even though a line or component ruptures only a small amount of water would be removed from the tank (< 3 ft) by a siphon block (see Chapter 5). Rupture of the reactor tank is virtually impossible since it is supported on the bottom by a reinforced concrete slab and the tank wall is buttressed by the surrounding the earth.

Criterion 15: Reactor Coolant System Design

The reactor tank is an open system and the maximum pressure in the primary system is due to the static head (about 23.5 ft). The primary cooling system, the secondary cooling system, and the purification system are pressurized by small capacity pumps.

The piping and valves in the primary and purification systems are PVC or aluminum, and of such size so as to provide adequate operating margins. The secondary system components are PVC and carbon steel. Chapter 5 describes the coolant system in detail.

Criterion 16: Containment Design

The structure surrounding the reactor constitutes a confinement building rather than providing absolute containment. Because of the low fission-product inventory, leakage from the structure can be tolerated.

Criterion 17: Electric Power Systems

The building (MEB) provides electrical power to the reactor console, continuous air monitor (CAM), and area radiation monitors (ARM), during normal reactor operations. With loss of building power, a battery backup system is available. THE UUTR is cooled by PASSIVE natural convection and there is no requirement to provide forced cooling flow for the removal of decay heat, so response time for the reactor operator to shutdown the reactor and confirm shutdown is adequate. The battery backup system provides twenty-four hours of power to the security system, CAM, and ARM.

University of Utah electrical maintenance crews maintain the primary power distribution system supplying commercial power to UUTR. Routine inspections of the systems are performed.

The UUTR can tolerate a total loss of electric power with no adverse effects on the safety of the facility. There is no electrical power (distribution) systems necessary to provide cooling to the UUTR during either normal or abnormal conditions (see Chapter 13).

Criterion 19: Control Room

In the event of an accident when operating procedures require shutdown of the reactor, occupancy of the control room is not necessary since the reactor has been shut down and experiments are terminated. Exposure levels derived from an accident's radiation sources would be significantly reduced in magnitude (due to the location of the control room with respect to the reactor room). Consequently, control room radiation levels will not exceed allowable tolerance levels; nevertheless, the UUTR Emergency Plan describes actions for mitigating accident situations, which require control room evacuation.

3.1.3 Protection and Reactivity Control Systems (Criteria 20-29)

Criterion 20: Protection System Functions

The UUTR Reactor Protection System has been designed to initiate automatic actions to assure that fuel design limits are not exceeded by anticipated operational occurrences or accident conditions. The automatic actions are initiated by two power channels and two fuel temperature channels. The Reactor Protective System automatically scrams the control rods when trip settings are exceeded. (See Chapter 7.) There are no other automatic actions required by UUTR systems to keep fuel temperature limits from being exceeded. The Reactor Protective System satisfies the intent of IEEE-323-1974 in the areas of redundancy, diversity, power-loss, fail-safe protection, isolation and surveillance.

Criterion 21: Protection System Reliability and Testability

The UUTR protection system is designed to be fail-safe. Any channel signal or functional loss that causes the channel to lose its ability to perform its intended function results in initiation of shutdown action. Protective action is manifested through several independent scram inputs arranged in series such that any event that interrupts current to the scram magnets results in shutdown of the reactor. Redundancy of channels is provided. In addition, a loss of any channel due to open circuit or loss-of-power will result in a scram. Scram action is, therefore, on a one-out-of-one basis. All instrumentation is provided with testing capability. The Reactor Protective System satisfies the intent of the IEEE-323-1974 standard. Additionally, all safety systems are tested during a reactor checkout prior to a run or control rod manipulation.

Criterion 22: Protection System Independence

The protective system satisfies the intent of IEEE-323-1974 "Criteria for Protective Systems for Nuclear Power Generating Stations." Protective functions are initiated through two independent power and two independent fuel temperature channels that provide a diversity of independent scram modes. Furthermore, the protective system is fail-safe upon loss of power.

The Reactor Protective System and the magnet power supply are, for the most part, physically and electrically isolated from the remainder of the control system. There is a separate conduit for each safety channel and one for the magnet power supply.

Criterion 23: Protection System Failure Modes

The reactor protective system is designed and constructed to be fail safe in the event of any failure of a safety channel. Failure of a safety channel will result in removal of power to the control rod magnets, dropping the control rods into the core. The reactor protective system contains no control functions. Therefore, loss of a protective function will not affect the operation of the reactor, such as initiating an uncontrolled reactivity insertion.

Criterion 24: Separation of Protection and Control Systems

The UUTR has four power indicating channels, two fuel temperature channels, and two water temperature channels. One of the power channels utilizes a fission chamber for startup count rates. This channel provides signals for both safety (source interlocks) and control action. The linear power channel utilizes a compensated ion chamber. This channel provides linear power for safety (scram) action as well as power monitoring capability. The third channel is a percent power channel. It is the primary safety channel with scram and power monitoring capabilities. The forth power monitoring channel (log channel) is not linked to a safety function. Fuel temperature is measured by thermocouples placed within the special instrumented fuel elements. The information from these channels is processed and displayed on the console independently, and connects directly to the safety system scram circuit. The ability of this configuration to meet the intent of protection system requirements for reliability, redundancy, and independence for TRIGA-type reactors has been accepted by the NRC.

Finally, the control and safety systems are fail-safe and will scram the reactor should they malfunction. No control or safety systems are required to maintain a safe shutdown condition.

Criterion 25: Protection System Requirement for Reactivity Control Malfunction

The UUTR Protection System is designed to assure that fuel temperature limits are not exceeded for any single malfunction of the reactivity control system. However, Chapter 13 shows that the simultaneous and accidental removal of all rods from the core at their normal drive speed will not exceed fuel temperature limits.

Criterion 26: Reactivity Control System Redundancy and Capability

The UUTR has three independent reactivity control rods: one shim rod, one regulating rod, and a safety rod. The shim and safety rods have nearly identical reactivity worths; therefore, for certain specialized applications, the role of these two rods is reversed. Both rods are capable of shutting the reactor down well below the shutdown margin. Each of the CONTROL rods has its own drive mechanism and control circuit and they can scram independently of each other. The regulating rod is used for fine power control and adjustment.

Upon receipt of a scram signal, all three rods are released from their drives and dropped into the core. Insertion of any two of the three rods ensures reactor shutdown.

Criterion 27: Combined Reactivity Control System Capability

Emergency core cooling is not required for the UUTR. Analyses (Chapter 13) have shown that the worst conditions resulting in instant loss of coolant do not cause fuel-element temperatures to reach the safety limit.

Total worth of the rods is more than adequate to maintain the core at a subcritical level even with the most reactive rod removed from the core.

Criterion 28: Reactivity Limits

No conceivable malfunction of the reactivity control systems could result in a reactivity accident worse than the conditions encountered during an accidental maximum-yield pulse, as outlined in Chapter 13. As shown in Chapter 13, neither continuous rod withdrawal nor loss of coolant will cause undue heating of the fuel. Identified accidents will not result in significant movement of adjacent fuel elements or otherwise disturb the core so as to add reactivity to the system.

Since the primary coolant system operates at atmospheric pressure, control-rod ejection is not a credible event. The shim rods, the regulating rod, and the safety rod cannot drop out of the core because the rods in the full down position are approximately one inch above the safety plate located near the bottom of the tank. Travel out of the core in the downward position is therefore eliminated.

Criterion 29: Protection Against Anticipated Operational Occurrences

The protection and reactivity control system satisfy all existing design standards. Periodic checks (i.e., startup, shutdown, and maintenance procedures) of all reactor protective system channels and reactivity control systems demonstrate that they will perform their intended function. If there was a loss of electrical power, the reactor would scram due to the loss of current to the control rod electromagnets. Because the reactor is cooled by natural convection, and there is no requirement to provide forced cooling flow for the removal of heat, there is sufficient time for the reactor operator to shutdown the reactor and confirm shutdown.

3.1.4 Fluid Systems (Criteria 30-46)

Criterion 30: Quality of Reactor Coolant Pressure Boundary

The reactor tank is open to the atmosphere and is subjected only to ambient conditions. All components containing primary coolant (i.e. reactor tank, primary coolant systems and the purification system) are constructed of PVC, aluminum, and stainless steel, using standard codes for quality control. There is no requirement for leak detection in the primary coolant or purification loop since no conceivable leak condition can result in the tank water level lowering more than approximately three feet.

Criterion 31: Fracture Prevention of Reactor Coolant Pressure Boundary

Since the coolant system is open to the atmosphere, no reactor coolant pressure boundary is required.

[redacted] The tank wall cannot be inspected by any means other than visual observation through the water from inside the tank. All piping, valves, meters, etc. associated with the primary water system are located in open spaces and are readily accessible for periodic inspections.

The UUTR operates at relatively low powers and temperatures. Because of the moderate fluence levels and low temperature factors, no significant change in material properties is expected.

Criterion 33: Reactor Coolant Makeup

The UUTR water purification system design includes a system for makeup of primary coolant water. This system is manually operated. (See Chapter 5.)

Criterion 34: Residual Heat Removal

Natural convection cooling is adequate to dissipate the heat from the core. Many years of operations with TRIGA reactors have shown that natural convection will provide adequate flow for the removal of heat after several hours of maximum steady-state operation. Furthermore, calculations performed for loss of coolant (see Chapter 13) show that fuel temperatures will not reach the safety limit even under loss-of-coolant conditions.

Based on the above, there is no requirement to provide a residual heat removal capability.

Criterion 35: Emergency Core Cooling System

An emergency core-cooling system is not required for the following reasons:

1. The system is not pressurized and does not operate at high coolant temperatures.
2. Natural convection cooling will adequately dissipate generated core heat.
3. If all water is lost for any reason, air cooling (as shown by analysis in Chapter 13) will satisfactorily dissipate heat and prevent exceeding the safety limits.

Criterion 36: Inspection of Emergency Core Cooling System

This criterion is not applicable.

Criterion 37: Testing of Emergency Core Cooling System

This criterion is not applicable.

Criterion 38: Containment Heat Removal

There are no systems, devices, equipment, experiments, etc., with sufficient stored energy to require a containment heat removal capability. This criterion is not applicable

Criterion 39: Inspection of Containment Heat Removal System

This criterion is not applicable.

Criterion 40: Testing of Containment Heat Removal System

This criterion is not applicable.

Criterion 41: Containment Atmosphere Cleanup

Post accident activities are not contingent upon maintaining the integrity of the building structure. Accident analyses (see Chapter 13) have shown that downwind doses would not exceed 10 CFR Part 20 or ANSI 15.7 guidelines in any credible accident. Routine operations result in two isotopes of concern being produced: Argon-41 in the reactor room and Nitrogen-16 from the irradiation of oxygen in the tank water. Analysis in Chapter 11 shows concentrations to be below ANSI 15.7 guidelines for accident situations and below 10 CFR Part 20 guidelines for normal operations.

Criterion 42: Inspection of Containment Atmosphere Cleanup Systems

This criterion is not applicable.

Criterion 43: Testing of Containment Atmosphere Cleanup Systems

This criterion is not applicable.

Criterion 44: Cooling Water

A coolant system is utilized to cool reactor tank water during normal operation of the reactor. The UUTR requires no auxiliary cooling system for cooling of reactor tank water upon shutdown.

Criterion 45: Inspection of Cooling Water System

Cooling equipment used in normal operation of the reactor is located either in the reactor room, with adequate space provided to permit inspection and testing of all components. Operation of the bulk coolant and cooling system is checked on a daily basis prior to reactor operation. During this checkout, the performance of each system is monitored with emphasis on system flow rates and temperature.

Criterion 46: Testing of Cooling Water System

UUTR reactor cooling systems are routinely checked, tested, and maintained.

3.1.5 Reactor Containment (Criterion 50-57)

Criterion 50: Containment Design Basis

Under the conditions of a loss of coolant, it is conceivable that the temperature at the reactor room could increase slightly due to heating of the air flowing through the core. However, since the building is not leak tight, it will not pressurize from the heating of the air.

Furthermore, there is no requirement from a radiological exposure viewpoint for a containment structure; hence, only a confinement structure exists at the site. In addition, there is no source of energy (from an accident) that would provide a significant driving force if no corrective action was taken.

Criterion 51: Fracture Prevention Boundary

The confinement structure (the reactor room) is a building with reinforced filled concrete walls and the ceiling is a steel and concrete load-bearing floor. The entire structure is exposed to only normal external environmental conditions and internal environmental conditions are maintained at regulated conditions.

The structure will not be subjected to significant internal pressures during normal operations. Postulated accident conditions cannot result in significant changes in the pressure differential due to the non-leak tightness of the structure.

Criterion 52: Capability for Containment and Leakage Rate Testing

This criterion is not applicable.

Criterion 53: Provisions for Containment Testing and Inspection

The reactor room confinement capability is checked on a daily basis prior to reactor operation and routinely throughout reactor operations. This check involves monitoring the pressure differentials between the reactor room and the surrounding areas. The reactor room exhaust recirculation system is checked monthly to confirm proper operation.

Criterion 54: Piping Systems Penetrating Containment

Piping systems that involve penetrations through the reactor building walls have no effect on the safety of operation; therefore, isolation, redundancy, and secondary containment of these systems are not required.

Criterion 55: Reactor Coolant Pressure Boundary Penetrating Containment

The reactor room was not designed nor constructed as a pressure containment structure, but does provide adequate airborne radioactivity confinement. As pointed out in the responses to previous criteria, there are no requirements for containment (or confinement) capabilities. The only systems that penetrate the reactor room are the ventilation system, primary coolant system, makeup water system, control and console monitoring cables, and the pneumatic transfer tube experimental facility.

Criterion 56: Primary Containment Isolation

Penetrations through the building walls have no effect on the safety of reactor operations; therefore, isolation systems are not required in the UUTR.

Criterion 57: Closed System Isolation Valves

The UUTR reactor building was designed to provide only confinement capability; isolation valves are not required.

3.1.6 Fuel Radioactivity Control (Criteria 60-64)

Criterion 60: Control of Release of Radioactive Materials to the Environment

There is no readily available path for liquid waste to be discharged directly to the environment. Liquids in the reactor room may result from spills, wash down of the floor, etc. These liquids are collected in a storage tank within the UUTR, analyzed for radioactivity, and disposed of accordingly (see Chapter 13).

Criterion 61: Fuel Storage and Handling and Radioactivity Control

The major concern relative to storage, handling, and control of radioactivity of irradiated fuel is shielding. All irradiated fuel elements are either stored in special racks (see Criterion 62) in the reactor tank or [redacted]. When fuel is stored in the reactor tanks, the water provides a minimum shield thickness of at least 18 ft. This amount of water also provides scavenging of any fission products should any escape from the fuel elements. Lead covers provide shielding for elements stored in the reactor room storage pits. Cooling is not required due to low burn up and no large decay heat source is present in the UUTR fuel. Irradiated fuel elements are handled either under water or with a cask. The elements are transferred one at a time so they are in a criticality-safe configuration (see Chapter 9).

Some spare, unirradiated, UUTR fuel elements may be stored in a criticality-safe configuration [redacted]. These elements require no special handling arrangements or radiation shields.

For some experiments, special core loading may be required. Fuel elements removed from the core can be placed in a criticality-safe fuel storage rack attached to the tank wall.

Criterion 62: Prevention of Criticality in Fuel Storage and Handling

Fuel-storage capability is provided by storage racks mounted in the tank [redacted]. The storage locations are criticality safe due to the geometry utilized and the limited quantity of fuel elements, which can be stored (see Chapter 9). Since only one fuel element can be handled at a time, handling does not present a criticality problem.

Criterion 63: Monitoring Fuel and Waste Storage

The reactor room and the UUTR fuel storage area radiation level are monitored with both an ARM system and a CAM system. No residual heat removal or temperature measuring capability is required for irradiated UUTR fuel elements. Fuel burn up is low; therefore, only a minimum decay heat source is present.

Criterion 64: Monitoring Radioactivity Releases

The radiation monitoring system for the UUTR consists of the ARM's and CAM's. ARM's monitor the reactor room and selected areas outside the reactor room for gamma activity.

The UUTR exhaust stack is equipped with a CAM, which provides continuous readings of radiation from Ar-41 and beta/gamma particulate released from the facility. This CAM has local readouts and alarms as well as remote readouts and alarms in the reactor control room. Actions initiated to reduce the release of radioactivity if the set points of this instrument are exceeded are discussed in Chapter 9 and Chapter 11. This CAM has local readouts and alarms as well as remote readouts and alarms in the reactor control room.

3.2 Meteorological Damage

The UUTR reactor core is protected from damage due to high winds or tornadoes by virtue of its below grade location and the thick reinforced concrete structure surrounding the reactor area. The superstructure of the UUTR has been designed for area wind loads.

3.3 Water Damage

As discussed in Chapter 2, flooding is not expected at the UUTR site. However, even if flooding occurred, reactor safety would not be an issue since the core is located in a water pool. There are no pipes in the UUTR facility capable of flooding the reactor room to the first floor level. Furthermore, the lowest elevation in the UUTR, the reactor room floor, contains a sump.

3.4 Seismic Damage

The UUTR site is in a UBC, Zone 3, risk area. (See Chapter 2.) The UUTR building, reactor foundation, shielding structure, reactor tank, and core support structure are designed in accordance with UBC Zone 3 requirements. Meeting these requirements should ensure that the reactor can be returned to operation without structural repairs following an earthquake likely to occur during the plant lifetime. Furthermore, failure of the reactor tank and loss of the coolant in the event of a very large earthquake has been considered in Chapter 13 and the consequences found acceptable from the standpoint of public safety. Seismic considerations applicable to the UUTR facility are discussed in Chapter 2 and 13.

3.5 Systems and Components

3.5.1 Classification of Structures, Components, and Systems

The UUTR reactor does not have structures, components, or systems that are important to safety in the same context as nuclear power plants. For the UUTR, a loss of the coolant systems, failure of the protection system, or any other credible accident does not have the potential for causing off-site exposure comparable to the guideline exposures of ANSI 15.7.

The UUTR facility does not have any structures, components, OR systems requiring a Category I classification. However, essential structures, components, and systems have been designed to withstand natural phenomena that may occur during the plant lifetime. These design considerations are discussed in the following subsections.

3.5.2 Systems-Quality-Group Classifications

Classification of the UUTR fluid systems into quality groups (in accordance with the Regulatory Guide 1.26 quality-group classification system) is considered inappropriate because these systems need not remain functional to ensure that the reactor can be maintained in a safe shutdown condition and to prevent the release of significant quantities of radioactive material to the environment.

3.5.3 Control Rod Drives

The control-rod-drive assemblies for all control rods are mounted on the reactor bridge structure. The drives are standard TRIGA drive mechanisms manufactured by General Atomics (GA). The mechanism consists of a stepping motor and reduction gear, a rack and pinion, an electromagnet and armature, a dashpot assembly, and a control-rod extension shaft. Rod-position data are obtained from potentiometers. Limit switches are provided to indicate the up and down positions of the magnet and the down position of the rod. The drive motor is of the stepping type and is instantly reversible. The nominal drive speed for the shim and the regulating rods is 30 in/min. The stepping motor speeds are adjustable with a maximum rod speed of removal of 1.016 cm/sec. Rod reactivity insertion accidents use this maximum rate. (See Chapter 13.)

During a scram, the control rod, rod extension, and magnet armature are detached from the electromagnet and drop by gravity. The dashpot assembly slows the rate of insertion near the bottom of the stroke to limit deceleration forces.

3.5.4 Core-Support Structure

The fuel elements, heavy water, and graphite assemblies are supported by the core-support structure. The UUTR grid plate has been designed to have a thickness and hole pattern identical to those of other TRIGA reactors with hexagonal grids (see Chapter 4).

3.5.5 Neutron Source

The startup source is Plutonium-Beryllium held in a triple encapsulated stainless steel container approximately the same size as a fuel element. The capsule is held in a container that positions the source at the edge of the core near the fission startup chamber, Chapter 4 gives a detailed description of the source capsule and holder.

3.5.6 Fuel Storage Assemblies

[redacted]

3.5.7 Beam-Tube Assemblies

Three beam tubes originate from the core shroud at approximately 120° apart. The beam tubes are described in Chapter 9.

Table 3.1.1
General Design Criteria and the UUTR- Protection by Multiple fission product barriers and Protection and Reactivity control systems

Criterion Number and Title	Compliance	Compliance not required	Conditional Non compliance	Conditional Compliance
1. Quality Standards and records	X			
2. Design Basis for Protections against natural phenomena	X			
3 Fire protection	X			
4. Environmental and Missile design basis	X			
5. Sharing of structures	X			
Protection by Multiple Fission Product Barriers				
10. Reactor design	X			
11. Reactor Inherent Protection	X			
12. Suppressions of Reactor Power Oscillations	X			
13. Instrumentation and Control	X			
14. Reactor Coolant Pressure Boundary	X			
15. Reactor Coolant system design	X			
16. Containment Design	X			
17. Electrical Power system	X			
18. Inspection and testing of electrical Power systems	X			
19. Control room	X			
Protection and Reactivity Control Systems				
20. Protection system function	X			
21. Protection system reliability and testability	X			
22. Protection system independence	X			
23. Protection system failure modes	X			
24. Separation of protection and control system	X			
25. Protection system Requirements for reactivity control malfunctions	X			
26. Reactivity control system redundancy	X			
27. Combined reactivity control system capability	X			
28. Reactivity limits	X			
29. Protection against anticipated operational occurrences	X			

Table 3.1.1

General Design Criteria and the UUTR- Fluid Systems and Reactor Containment

Criterion Number and Title	Compliance	Compliance not required	Conditional Non compliance	Conditional Compliance
Fluid Systems				
30. Quality of reactor coolant pressure boundary		X		
31. Fracture Prevention of Reactor coolant pressure boundary		X		
32. Inspection of reactor coolant pressure boundary	X			
33. Reactor coolant makeup	X			
34. Residual heat removal		X		
35. Emergency core cooling		X		
36. Inspection of emergency core cooling		X		
37. Testing of Emergency core cooling system		X		
38. Containment heat removal	X			
39. Inspection of Containment heat removal		X		
40. Testing of Containment heat removal		X		
41. Containment of Atmosphere Cleanup	X			
42. Inspection of Containment of Atmosphere Cleanup		X		
43. Testing of Containment atmosphere Cleanup system		X		
44. Cooling water	X			
45. Inspection of cooling waster system				X
46. Testing of Cooling water system				X
Reactor Containment				
50. Containment design basis	X			
51. Fracture prevention of Containment Pressure boundary	X			
52. Capability for Containment Leakage rate testing		X		
53. Provisions for containment testing and inspection		X		
54. System Penetrating Containment		X		
55. Reactor Coolant Pressure boundary penetrating containment				X
56. Primary containment isolation		X		

57. Closed system Isolation Valves		X		

Table 3.1.1

General Design Criteria and the UUTR- Section 3 of 3

Criterion Number and Title	Compliance	Compliance not required	Conditional Non compliance	Conditional Compliance
Fuel and Radioactivity Control				
60. Control of Releases of Radioactive Materials to the Environment	X			
61. Fuel Storage and Handling and Radioactive control	X			
62. Prevention of Criticality in Fuel storage and Handling	X			
63. Monitoring Fuel and Waste storage	X			
64. Monitoring radioactivity Releases	X			

Chapter 4

Reactor Core Description

4 DESCRIPTION OF THE UUTR REACTOR 1

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List of Figures and Tables

(Figures and Tables are numbered according to the sections that refer to them.)

4 Description of the UUTR reactor

Chapter 4 contains a description of the reactor. Specifically addressed are the reactor core structures, fuel, experimental facilities, and instrumentation. Some of the topics are discussed in other chapters of this safety analysis report.

4.1 Reactor Core

4.1.1 Grid Plate

The upper and lower grid plates were constructed from drawings and specifications supplied by Gulf Energy and Environmental Systems Inc. Figure 4.1.1 shows a top view of the TRIGA reactor's core. The upper grid plate is 3/4-inch thick, type 6061-T6 aluminum. It laterally positions the fuel elements, control rods, irradiation facilities, and neutron source, heavy water and graphite reflectors. The plate carries no vertical load other than its own weight. It is mounted to six side plates, each 1/4 inch thick, that are secured to the bottom grid plate. The upper grid plate has 127 locations for fuel elements (or moderator rods, control rods, etc.). These 127 holes are in six hexagonal rings around the center hole and are 1.505 (+0.005, -0.000) inch in diameter with a hexagonal lattice pitch of 1.72 inches. Cooling water passes through the top plate by means of the clearance provided by the differential area between the triangular spacer blocks on the top of each fuel element and the round holes in the grid plate. There are also several small diameter flux-wire insertion holes in the top grid plate in the interstices between the fuel element holes. These holes line up with identical holes in the bottom grid plate. A 7 central fuel element cutout is provided for A & B ring positions that can accommodate a 4.215-inch diameter experiment with a cross-section area of 13.96 square inches.

The lower grid plate provides lateral positioning and supports the entire weight of the core. It is also type 6061-T6 aluminum, and is 3/4 inches thick. It rests on six legs (pads), six inches from the base of the aluminum reactor tank. The 1/4-inch-thick hexagonal aluminum shroud plates for the core are attached to the lower grid plate. The lower grid plate also contains 127 holes for fuel elements, but each hole is 0.25 inches in diameter with a 5/8-inch diameter counter sink. The tapered fuel element ends fit securely into these holes. Cooling water cannot pass through the fuel element holes in the lower grid plates. However, cooling water does flow through 5/8-inch diameter holes in the interstices between each fuel element position in the lower grid plate. The fuel insertion holes in the lower grid plate line up with the similar holes in the top grid plate. Since the control rods contain no followers, any fuel element position may also accommodate a control rod.

4.1.2 Moderated Fuel Elements

The aluminum and stainless steel clad fuel elements and the optional graphite or heavy water loaded reflector elements all have similar outside dimensions. [redacted] The fuel is a solid, homogeneous mixture of uranium-zirconium hydride alloy containing between 8.5% and 30% by weight of uranium enriched to < 20% in U-235. The hydrogen to zirconium ratio is approximately 1.0. Each aluminum-clad element is clad with 0.030-inch-thick aluminum, and the stainless steel clad elements have a clad thickness of 0.020 inches. [redacted] Each end has 4-inch-long sections of graphite. Aluminum end fixtures are on both ends of the fuel element as shown. The upper fixture contains a knob for grasping with the fuel-handling tool.

Fuel elements will be operating well below the design basis limits, no stress-associated failure of these fuel elements is expected. (See appendices A and B) If water should leak into the graphite or heavy water elements, the effect would be a loss of reactivity.

The TRIGA reactor system is well known for its conservative design. The stability of this reactor type has been proven through calculation as well as through tests performed with the many TRIGA reactors in operation worldwide. The University of Utah's TRIGA reactor has the same inherent stability that has been demonstrated on other TRIGA systems. The stability of the TRIGA type reactor stems from the prompt negative temperature coefficient of the U-ZrH fuel in conjunction with a suitable neutron thermalizing material. A review of the reactivity worth of the reactor core indicates that no single item listed can produce a step reactivity insertion greater than that afforded by routine pulse operation at which other TRIGAs have been operated. In the pulsed mode of operation, a step insertion of 2.1% Δk/k ($3.00) could result in a reactor peak power of about 2000 MW with a prompt reactor period of 2.8 msec, an energy release of 26 MW-sec and associated peak fuel temperatures below 650°C. At this temperature, some dissociation of hydrogen from the ZrH molecule occurs and a temporary phase transition in the ZrH may take place. However, the energy release in this pulse is far below that conducted as test pulses on the advanced TRIGA prototype reactor in which 3.5% Δk/k ($5.00) is inserted in a step. These considerations show that the

UUTR core can withstand any single accident caused by sudden insertion of all of its available excess reactivity. Furthermore, the University of Utah TRIGA reactor will not be operated in a pulsing mode.

In the University of Utah's TRIGA reactor, two types of fuel are used: "low hydride" (aluminum fuel elements) and "high hydride" (stainless steel fuel elements). The reactor fuel temperature limits are dictated by the internal pressure in the fuel, which is caused by a phase change of the ZrH_X, fission gas buildup, and chemical reactions.

The performance of the "low hydride" fuel (x in ZrH_X less than 1.5) is dictated by the characteristics of the ZrH_X ceramic fuel matrix. As shown in Figure 4.1.3 A, the "low hydride" fuel undergoes a solid-solid phase transformation at a threshold temperature of 530°C. Therefore, 530°C has been established as the upper limit for the "low hydride" fuel.

Equilibrium Hydrogen

At 530 °C, the equilibrium hydrogen pressure for the fuel is less than 689 Pa (0.01 psi), as shown in Figure 4.1.3 B, and distortion of the fuel by phase transformations will not occur. Thus, the cladding integrity is assured since the stress at 689 Pa on the 0.030-inch-thick aluminum cladding is only 16.09 kPa and 24.12 kPa for the 0.020-inch stainless steel cladding.

Fission Gas Buildup

With regard to fission gas release, the U-ZrH fuel has been shown to retain a large fraction of even the gaseous fission products. Assuming that all of the noble gases escape from the fuel matrix, 10% burn-up of the fuel loading in an element will create approximately 0.003 moles of noble gas atoms. Assuming that the noble gas atoms leave the fuel matrix and accumulate in 10 cm^3 of effective plenum area, the pressure and stress created on the cladding can be determined. Table 4.1.2 contains the stress created from the fission gas pressure for a range of temperatures. The use of 10 cm^3 as the characteristic volume is realistic since a large fraction of the noble gases will not escape from the matrix. From Table 4.1.2 it is seen that internal pressure and resulting stresses are well below the yield strengths of the fuel cladding.

Table 4.1.2
Pressure on the Cladding due to Fission Product Gases

Temperature °C	Pressure		Tangential stress	
	psi	kPa	psi	kPa
20	19.2	133	273	1885
100	24.5	169	348	2400
200	31.1	214	442	3044
300	37.6	259	534	3687
400	44.2	304	628	4330
500	50.8	350	722	4973
600	57.3	395	815	5617

Typical stress vs. temperature data (Nuclear Engineering Handbook, H. E. Thorington, Editor, 1958) for aluminum and aluminum alloys show that the creep rate under these conditions is quite tolerable, less than 10^{-3}% per hour. Thus, 460°C is an acceptable limit for clad temperature with a fuel element burn-up of as high as 10%. Beyond the 10% burn-up level, the reactivity penalty will be intolerably high, and generally precludes use of more than a few fuel elements to this level of burn-up. Fuel element logs are kept in which approximate burn-up is periodically estimated.

Chemical Reactions

Among the chemical properties of U-ZrH, the reaction rate of the material with water that might leak in through the cladding is of particular interest. Since the hydrating of zirconium is an exothermic reaction, water will react more readily with zirconium than with zirconium hydride. Hence, the water reaction is unlikely to occur with ZrH. The reaction is more likely to occur with uranium. Experiments carried out at Gulf General Atomic Incorporated studied the quenching with water of both powder and solid specimens of U-ZrH after heating to as high as 850°C. A relatively low chemical reactivity was observed with both water and air. Thus, a water leak in the cladding will not result in a rapid chemical reaction, and gases produced in the slow chemical reaction would probably dissipate slowly throughout the fuel.

In summary, limits of 800°C and 460°C, for respectively stainless steel and aluminum fuel element matrices, will assure sufficiently low internal pressures and stress for the cladding to preserve fuel element integrity and prevent any breach in the cladding. The design basis limit (safety limit for reactor operation) is that the temperature of a stainless steel "high hydride" fuel element not exceeds 1000°C, and an aluminum clad "low hydride" full element shall not exceed 530°C under any conditions of operation. Furthermore, if the cladding should fail for other reasons, these same temperature limits will assure that any chemical reaction occurring with the fuel will be minor.

4.1.3 Core Arrangements

The active core has the approximate shape of a right hexagon. Fuel element spacing provides that 33% of the cross-sectional core area in the lattice is water. The extra fuel element spaces on the outside of the active core allow for the insertion of reflector elements. The operational loading with a cold, clean k-excess of 2.25%Δk/k ($3.21) is expected to be approximately 75 elements, if these are reflected with one row of reflector elements.

4.1.4 Core Reflection

The reflector will generally consist of a single row of graphite or heavy water reflector elements. These are such that the equivalent volume fraction of the first 1.6 inches of reflector is approximately 33% water, 62% graphite or heavy water, and 5% aluminum. The remaining part of the reflector is water, plus the 0.25-inch-thick aluminum core shroud.

4.1.5 Startup Source

[redacted] A Source Interlock indicator light visible to the operator indicates if the source is in or out of the core. Also the Startup Channel's fission chamber is located next to the Startup Source, and this channel's counts will show the presence of the Startup Source. In addition, there is a video camera positioned above the reactor viewing the core with a monitor in the control room. Source position can be verified from this monitor.

A fission counter is used with a transistorized linear amplifier. The meaningful count rates range from approximately 10^{-3} W to about 2 W (source level). These source levels are estimated to give count rates of about 5 counts/second and 10,000 counts/second, respectively. An interlock circuit is used to prevent rod withdrawal unless the source count level is above the required minimum value of at least 2 counts/second.

4.2 Reactor Pool "TANK"

[redacted] All outer surfaces of the latter tank that come into contact with soil are painted to inhibit corrosion. The inner surface is painted with epoxy. The two-foot-space between the liners is filled with tamped sand to the concrete pad base. The bottom of the aluminum tank is a welded sheet of aluminum material the same as the side walls (6061-T6 aluminum). Both tanks are water tight, and welds on the inner aluminum tank have been made to meet ASME unfired pressure vessel standards.

Since the bottom of the aluminum tank rests on a reinforced concrete pad that forms the bottom seal for the outer steel liner, the maximum stress at the bottom of the aluminum tank is the hoop stress at the bottom weld. Under a hydrostatic head of 24 feet of water, the hoop stress at the bottom of the inner tank is 24.1 MPa (3500 psig), while the hoop stress on the outer steel tank, if subject to the same head, would be 68.9 MPa (l0,000 psig), both well below the yield stresses of 246 MPa and 240 MPa, respectively, for these materials. [redacted] The general design of the tank and shield are shown in Figure 3.5. This design provides for shielding of the neutron flux from the earth by at least 2 feet of water in the tank and 2 feet of sand surrounding it. Approximately 20 feet of water and/or concrete and/or sand and earth above and to the side of the core provides the necessary shielding for personnel.

The reactor system contains provision for three diagonally directed beam tubes between the reactor core and the reactor room floor. Each tube is composed of two sections aligned along a common axis. The top tube section will be a 1-foot-diameter tube between the reactor floor and the wall of aluminum reactor tank. This tube will not penetrate the aluminum tank but will be sealed at the end where it butts against the tank. The upper beam tube will be adequately shielded with inner bags containing sand and capped at the reactor floor level with a 1/8-inch-thick steel cap for security, except when in use to extract a neutron beam for experimental purposes.

4.3 Fuel Storage

A fuel element storage rack is available for storage of new and used fuel elements around the inside perimeter of the tank. As fuel elements are burned up, they can be moved from the core to the side of the tank and replaced with new fuel elements to maintain the proper reactivity of the core. As the used fuel elements cool they can be moved from the perimeter storage rack to a storage pit.

[redacted] The crane travels on an overhead boom and is accessible to any location in the reactor room. The crane has a 4000-pound (2 ton) capacity and is locally controlled from a pendant box.

4.4 Fuel Handling Tools

A fuel element-handling tool is available for the movement of fuel elements in the core.

4.5 In-Pool Instrumentation

One compensated and two uncompensated ion chambers, a fission chamber, two instrumented fuel, are located in the water approximately on the core axial mid-plane on the outer perimeter of the core-shroud assembly. Chapter 7 contains a description of the instrumentation.

4.5.1 Fission Counter

A fission counter is used with a transistorized linear amplifier and preamplifier. The meaningful count rates range from approximately 10^{-3} W to about 2 W (source level). These source levels are estimated to give count rates of about 5 counts/second and 10,000 counts/second).

4.5.2 Power Monitoring Channel Detectors

Two power-level scram channels are available using one compensated and one uncompensated ion chamber to feed the transistorized power-level scram amplifiers. Additionally, these channels provide power-level readings at full reactor power (250 kW).

4.5.3 Temperature Sensors

Temperature sensors all consist of K-type thermocouples. There are two sensors in the pool to measure bulk water temperature, and one is located in a fuel element to monitor the fuel temperature. A sensor is also located on the effluent side of the heat exchanger to monitor the efficiency of the cooling system.

4.5.4 Water Level Sensors and Locations

There are two water level sensors; one is a mechanical plunger that activates a scram circuit when the plunger drops far enough to contact a limit switch, and the other is an ultrasonic detector that gives a level indication to the operator at the control console

4.5.6 Period Monitoring Channel

The signal for the period monitoring channel is provided from the period amplifier of the Log-N channel. The period amplifier is a General Atomic transistorized model AP-130 having a range of -32 seconds to +4 seconds. The period meter is located on the control console in the control room.

4.6 Safety System

4.6.1 General Features

If a situation occurs that could result in operation outside allowable power ranges or if certain safety requirements are not met, the safety system initiates a scram by cutting power to the electromagnets that hold the control rods to the drive mechanisms. This results in the control rods dropping into the core under the force of gravity, which shuts down the reactor.

4.6.2 Scram Functions and Set Points

The safety system functions and the associated set points are shown in Table 4.6.2.

Table 4.6.2
Safety system set-points

Fuel temperature	350 °C
Linear power channel	100% full scale
Percent Power Channel	110% of licensed power (275 kW)
Low level	1 mr/hr
High level	10 mr/hr

4.6.3 Alarm and Annunciator System

In the event of an unusual event, or emergency situation, the reactor operator will shut down or scram the reactor as is deemed necessary and then activate an alarm that signals the evacuation of the reactor room and general area. The switch for the alarm is located in 1205 (see Figure 2.9).

4.7 Area Radiation Monitors

4.7.1 General Features and Purpose

An area radiation monitoring system (ARMS) includes detectors at four locations. The object of these locations is to monitor the most likely positions of release to uncontrolled areas.

4.7.2 Detectors and Location

Two detectors are positioned in the reactor room to supply information on radiation levels in the facility. One is positioned at the very top of the reactor pool to detect the radiation at pool level. One is positioned on the ceiling above the reactor to monitor the conditions of the room and for persons on those levels above the reactor (floors 2 and 3). A third detector is mounted in the ventilation ducts of the laboratory to determine the radiation leaving the building through the ventilation system. A fourth area radiation monitor is located in the counting laboratory.

4.7.3 Readouts, Indicators, and Alarms

All four of the ARMS are monitored at the main control console and activate an alarm if the radiation level is above the set point. The emergency ventilation system is operated by the detector at the top of the reactor pool. The system provides an audiovisual alert and alarm levels. When the levels are exceeded, the emergency ventilation system will be activated.

4.8 Experimental Facilities

4.8.1 Central Irradiation Facility

The central irradiation facility is located in the central fuel pin position. A special tube has been constructed to accommodate samples and can be placed in the central fuel pin position by means of a cable. The dimensions of this assembly are the same as a fuel pin.

4.8.2 Beam Ports

The reactor system contains provisions for three diagonally directed beam tubes between the reactor core and the reactor room floor. Each tube will be composed of two sections aligned along a common axis. The top tube section will be a 1-foot-diameter tube between the reactor floor and the wall of aluminum reactor tank. This tube will not penetrate the aluminum tank, but will be sealed at the end where it butts against the tank. The upper beam tube will be adequately shielded with sand and capped at the reactor floor level with a 1/8-inch-thick steel cap for security. Currently there are no plans to open these beam ports

4.8.3 Pneumatic Sample Transfer Facility

A pneumatic sample transfer (PST) system may be operated within a 1.5 inch O. D. tube. It will be driven by helium gas. The PST will have a slight curve in its tube such that direct streaming of neutrons from the core to the surface of the pool cannot occur. The PTS system is controlled by the reactor operator. Additionally the terminus is locked when not in use. (Maddock T. 2005)

4.8.4 Fast Neutron Irradiation

The fast neutron irradiator is constructed in two pieces the stand and the box. The irradiator is designed to provide a sample exposure to neutrons with minimal moderation and minimal gamma dose. The irradiator is located outside of the grid structure and does not interfere with fuel cooling channels. The reactivity associated with the irradiator cannot result in a prompt critical condition if the device is accidentally separated for the core or flooded by water.

4.8.5 Dry Tube

A dry tube thermal irradiator is available for use in a trapezoidal shaped D_2O tank attached to the side of the core. The samples are inserted into the irradiator through a curved PVC tube from the top of the reactor to the top of the irradiator.

4.9 References

a. Gere, and Timoshenko, *Mechanics of Materials,* Brooks/Cole Engineering, Boston, 1984, Page 310.

b. GA-9064, "Safety Analysis Report for Torrey Pines TRIGA Mark III Reactor," Docket Number 50-227.

c. GA-471, "Technical Foundations of the TRIGA," Aug. 1958, pages 63-72.

d. GA-2070, "General Specifications and Description of the TRIGA Mark I Reactor," Aug. 1962, pages 12-13, 50-52.

e. GA-4339, "TRIGA Mark III Reactor Description," Dec. 1963, Pages 2-2, 3-1.

f. Maddock T.L. The Pneumatic Transfer System: An experimental facility, Master Thesis Nuclear Engineering, University of Utah, 2005

g. University of Utah Safety Analysis Report 1985

h. University of Utah Safety Analysis Report 1975

[redacted]

i. [redacted]

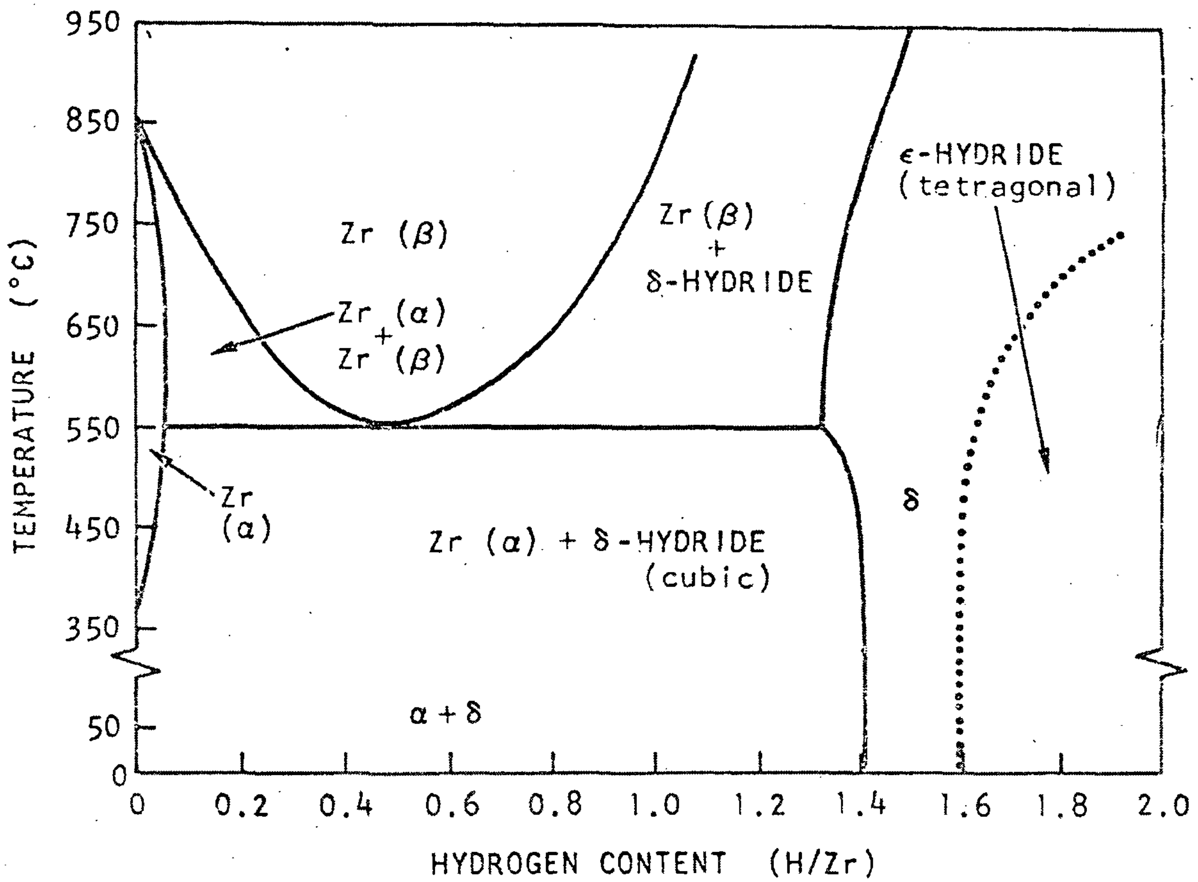

Figure 4.1.3A
Phase Diagram for Zr-H

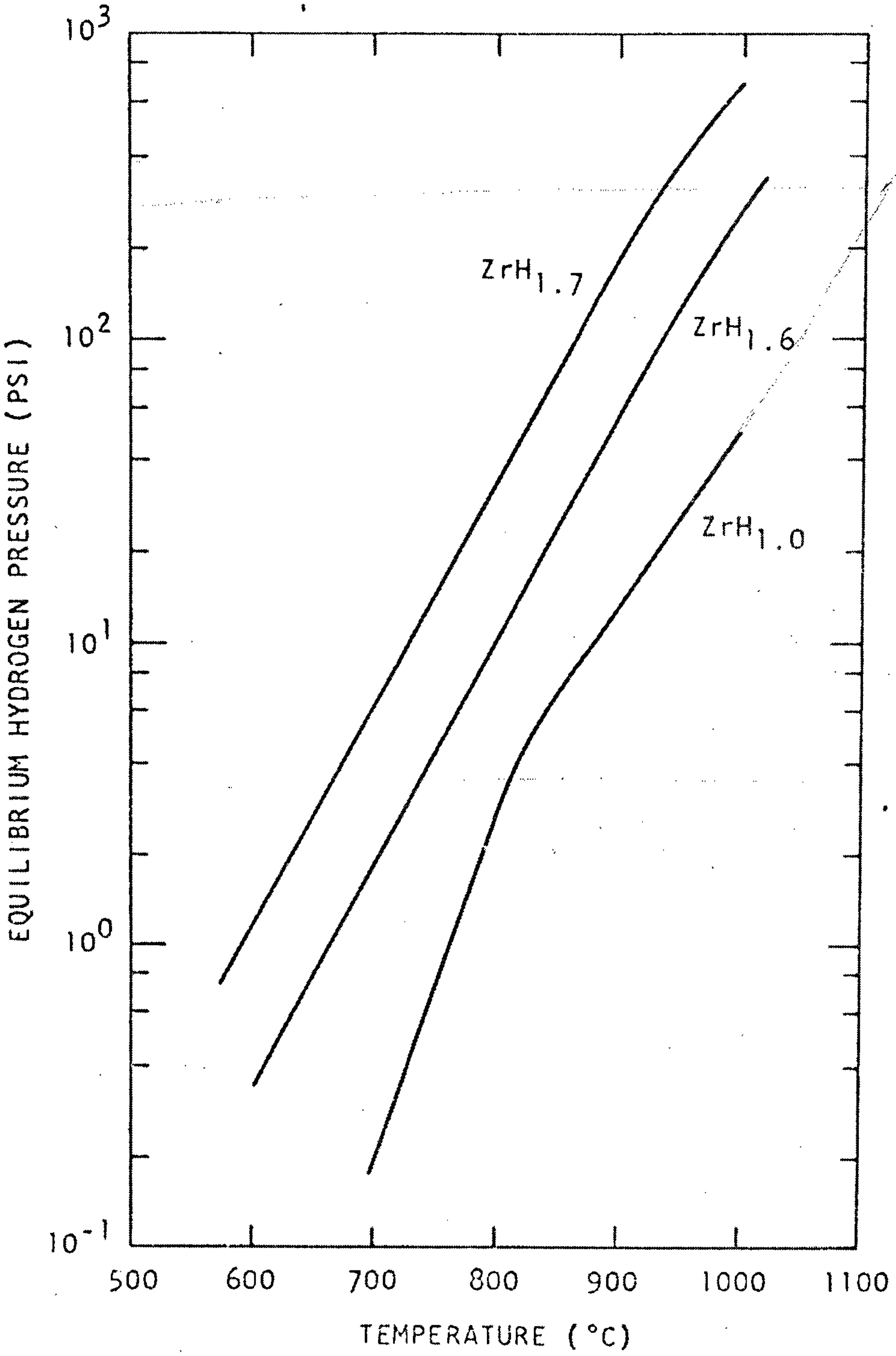

Figure 4.1.3B
Equilibrium Hydrogen Pressures over ZrH_x vs. Temperature

Chapter 5

Reactor Coolant Systems

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5 REACTOR COOLANT SYSTEMS

5.1 Summary Description

The UUTR is a natural convection water-cooled pool type reactor. The reactor pool is open to the atmosphere, while the secondary coolant system is a R134A based shell and tube heat exchanger. This secondary coolant system can dissipate 25 kW of heat.

The reactor core is positioned at the bottom of an open aluminum tank ~8 ft. diameter by 24 ft deep. The tank contains approximately 8,000 gallons of high-purity water and the core and fuel are clearly visible from the top. More than 20 ft of water over the top of the reactor core provides biological shielding for personnel in the reactor room. [redacted].

Tank materials, welding procedures, and welder qualifications were performed in accordance with the ASME code. The integrity of tank weld joints have been verified by dye penetrant checking and leak and hydrostatic testing.

5.2 Primary Coolant System

The reactor core is cooled by natural circulation of the reactor tank water. This tank water is the primary cooling system of the UUTR. The tank water temperature is maintained below 104 °F by limiting the run time at full power. A secondary cooling system is available to maintain this water temperature limit.

The natural convective flow of the primary cooling system can remove 250 kW of heat from the reactor fuel. An administrative limit of 104°F has been imposed on the bulk water temperature primarily to prevent possible skin scalding. Furthermore, at 140 °F the ion exchange resins can begin to degrade.

Data on the system parameters are displayed on the console. Visual and audible alarms are also located on the reactor control console, and are activated if the tank water level drops below preset limits. There are also tank level indicators in the control room, and on the control rod support structure in the reactor room.

All system components that contact the primary water are made from PVC, aluminum, or stainless steel.

5.3 Secondary Cooling System

The secondary cooling system is capable of continually removing 25 kW of heat from the primary system. The system circulates approximately 15 gpm of water from the reactor tank, through the primary-to-secondary shell and tube heat exchanger, to a R134a refrigeration system. The pressure of the secondary system is maintained at a higher level than the primary system to prevent cross contamination of secondary water should a leak develop in the heat exchanger.

To safeguard the shell and tube heat exchanger and prevent possible ice buildup and expansion, the refrigeration system is monitored during operation. The circulation pump for the primary coolant continues to operate while the refrigeration system is shut off.

5.4 Primary Coolant Cleanup System

The reactor water purification systems maintain the primary water purity and optical clarity. The system deionizes and demineralizes the water to maintain coolant water purity. A pre-filter is used to remove particulate matter prior to the water entering the deionizers.

A deionizing resin bed is supplied from the secondary cooling system (outlet of the heat exchanger) at a nominal flow rate of five gallons per minute (5 gpm). The resin bed is a fiberglass canister of mixed-bed resin.

Two independent conductivity cells are used to measure the conductivity of water entering the resin bed and the conductivity of the water exiting the resin beds subsequent to entering the reactor tank. There are readouts of the conductivity located on the reactor console.

5.5 Primary Coolant Makeup Water System

Makeup water to the primary coolant system is supplied from the Merrill Engineering Buildings culinary water system. This water is passed through a mixed resin bed prior to entering the tank. This mixed resin bed remains non-radioactive as it processes building water only. This extends the life of the primary coolant cleanup system. The outlet flow of the makeup system discharges to the purification system.

5.6 Auxiliary Systems Using Primary Coolant

As discussed above the primary coolant is the biological shield for the UUTR core. Thus, the temperature limits of 104 °F are specified for personnel protection.

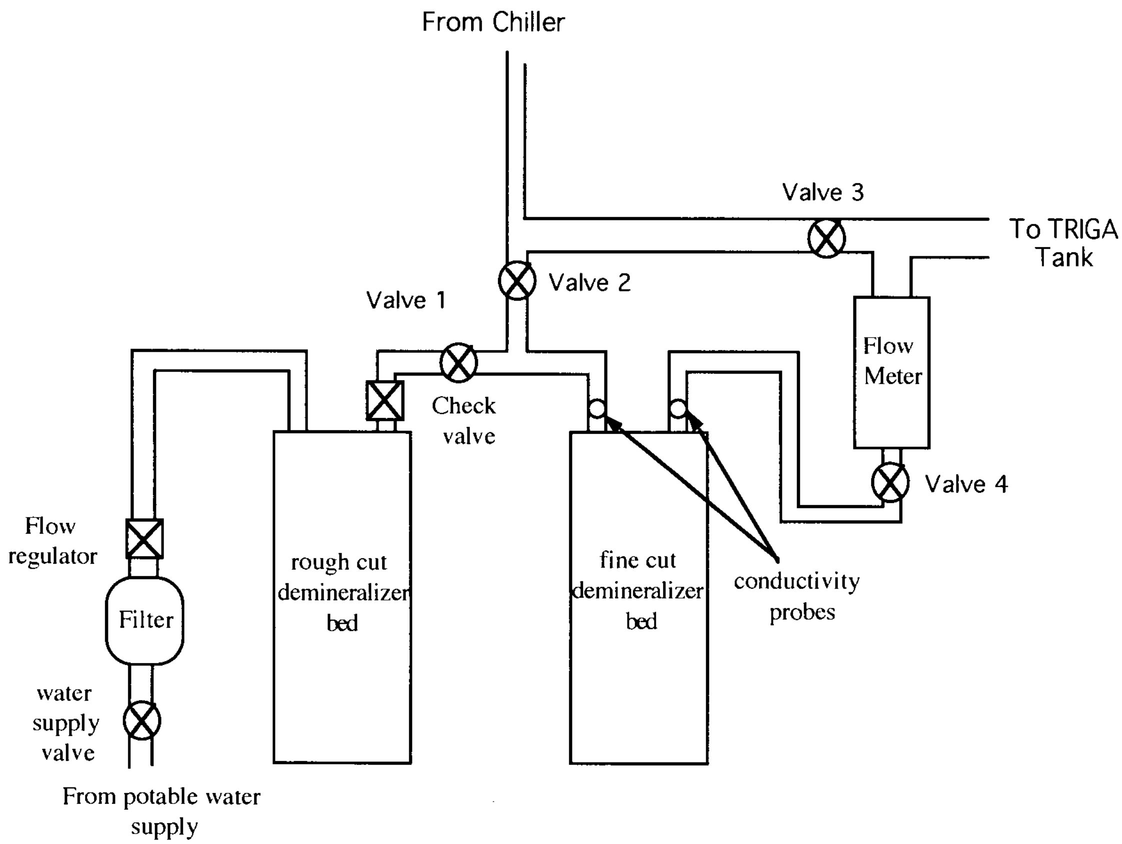

Figure 5.4 A

Primary coolant clean-up system

Chapter 6

Engineered Safety Features

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6 ENGINEERED SAFETY FEATURES

6.1 Summary Description

The UUTR reactor room employs a confinement type Engineered Safety System supported by a ventilation system that maintains the reactor room at a negative pressure differential state with respect to adjoining areas surrounding the reactor room. This system operates in two modes, viz., normal operation, and emergency operation. In both modes of operation, this system forces exhaust air through radiation monitors and HEPA filters. [redacted]. Below is a more detailed description of this system. Chapter 13 contains postulated situations that would be addressed by these engineered safety features.

6.2 Detailed Description of Engineered Safety Systems

The UUTR has a confinement type Engineered Safety System, no credit has been taken on any safety analysis for containment or emergency core cooling types of Engineered Safety Systems.

6.2.1 Confinement

The Engineered Safety Feature of the UUTR is a confinement system that meets the definition of confinement by having a robust enclosure that limits effluent exchange to all designed pathways. This pathway is a controlled and monitored release through HEPA filters, and isolation dampers. This system maintains a high negative pressure and is routinely leak tested. Under emergency conditions the isolation dampers increase this negative pressure as desired. Additionally, the UUTR facility is relatively robust against external events. [redacted]

6.2.1.1 Normal Operation

During normal operations the UUTR ventilation system operates 24 hours a day providing a negative pressure difference. Prior to every run this pressure differential is verified and documented.

6.2.1.2 Emergency Operation

If any of the three Area Radiation Monitors (ARM's) reads a radiation level greater than the preset 10 mR/hr, then the supply damper's set points are tripped and the dampers close. With the dampers closed, the reactor room draws a negative pressure greater than 0.02 inches of water. This attainment of this pressure differential in emergency mode is verified on a monthly basis.

6.2.2 Containment

No credit for the existence of a containment type Engineered Safety Feature has been taken in any of the safety analyses documented for the UUTR since a containment system is not required for the UUTR. See in Chapter 13.

6.2.3 Emergency Core Cooling System

An emergency core cooling system is not required for the UUTR. See Chapter 13. However, in the event of rapid and major loss of coolant, water from the building water supply, if operating, could be delivered to the reactor core from the water supply within the reactor room.

Chapter 7

Instrumentation And Control Systems

7 SUMMARY DESCRIPTION 7-1

List of Figures and Tables

(Figures and Tables are numbered according to the sections that refer to them.)

7 Summary Description

The types and operational characteristics of the instrumentation used on this reactor are analyzed in order to demonstrate that these instruments meet all design criteria for proper operation and, more importantly, safety of operation. The UUTR is currently operated in a steady state mode only; therefore, the additional restrictions applying to pulse or square wave operation are not required for this reactor. The two main items used in the reactor's control system are the fuel and control rods. The design bases for and functional characteristics of the reactor core and its components are discussed in Chapter 4 University of Utah TRIGA Reactor. Chapter 7 specifically addresses the design parameters of the instrumentation that monitors and controls conditions in the reactor core that ensure system components perform safely. The conditions monitored and controlled are: pool water level and chemistry and temperature, reactor power level, control rod position, fuel temperature console design, ventilation system and area radiation monitoring. Radiation monitoring will be addressed again in Chapter 11 Radiation Protection Program and Waste Management.

7.1 Pool Water

The major components of the pool water instrumentation include the following sub-systems: Water Level Scram, Water Temperature Function, Water pH Indicator, and Water Conductivity Indicators. These components exist so that the safety and long life of the reactor components are ensured for the life of the reactor.

7.1.1 Design Criteria

Several design criteria relate specifically to the coolant system for the UUTR. The tank water level is continuously monitored to ensure that the bulk shielding is always in place. The water quality in the tank is maintained such that reactor components (i.e. fuel, control rods, core structure) do not corrode or have mineral buildup.

Technical Specification 3.8, 4.5 and 5.7 require that the reactor shall not be operated unless there is 18 feet above the reactor core. There shall be a system in place to detect when the pool water level drops below 3 feet below the reactor tank top (18 feet above the reactor core). The conductivity of the pool water shall be no higher than $5 \cdot 10^{-6}$ mhos/cm.

The pH of the pool water shall be between 5.0 and 8.0. The conductivity and pH of the primary coolant water shall be measured monthly.

7.1.2 System Description

To ensure these specifications are met the pool is equipped with two water level meters, two conductivity probes the water cleaning/cooling loop, and a temperature and pH probe in the pool. All of these instruments have readouts on the console with the exception of one of the water level monitors. There are two water level meters in the UUTR pool. The pool water level scram function is tied to a float valve type sensor, while the console has an ultra sonic type sensor for precise measurements. An auxiliary system for pool water is a paddle wheel flow meter in the water loop.

7.1.2.1 Water Level Meter

The reactor tank water level safety system is a float valve. This float trips the SCRAM function when the water level drops 24 inches from the top of the tank. The reactor tank water level safety system tested prior to each reactor run and monthly. A second water level meter presents the reactor operator with a relatively precise measurement of the distance from the surface of the tank water to the top of the tank. This secondary meter is an ultrasonic distance measuring unit which is hung by a bracket from one of the beams supporting the rod drives. The ultrasonic level actuates an audible alarm if the water drops to 15 inches from the top of the tank.

7.1.2.2 Water Conductivity Meter

Two temperature compensated pure water conductivity probes are installed in the circulation system. The probes are specifically designed to measure very low conductivity solutions (like pure water) and therefore have a large surface area with a narrow space between the electrodes. One is in the line before the demineralizer beds (the pre probe) and one is in the line after the beds (the post probe). Due to the narrow spacing between their electrodes, these probes will only give correct readings with a reasonable water flow impinging on them. If the circulation plumbing is changed, these probes must remain in the water flow channel and not in a stagnant line.

7.1.2.3 pH Meter

The pH meter is connected to the high output impedance pH probe with a BNC terminated coaxial cable. An RTD is physically strapped to the probe. This temperature sensor provides temperature compensation information to the pH display. The probe is replaced whenever the reading does not change over the course of 10 minutes, or if it can no longer be calibrated using manufacturer recommended procedure.

7.1.2.4 Thermocouples

Pool water temperature is monitored by a K type thermocouple with a useful range of 0 to 100 °C and accuracy of ±1 °C.

7.1.2.5 Water Flow Meter

A paddle-wheel type water flow sensor is located on the circulation pump outlet line where the pump crosses over the top of the north edge of the reactor tank. The output of this sensor is an AC voltage; amplitude and frequency increase as the flow rate increases. For exact specifications, see the manual for this piece of instrumentation in the equipment filing cabinet, located in the UUTR Control Room.

7.1.3 System Performance Analysis

Thermal and hydraulic calculations performed by the vendor indicate that TRIGA fuel may be safely used to power levels of 2.0 MW with natural convection cooling. A water level limit of 18 feet above the top of the reactor core ensures that there is sufficient coolant for natural convection cooling. This amount of water also serves as sufficient radiation shielding when the reactor is operated at 250 kW. Estimated radiation level at the surface of the pool with 20 ft of water above the core is 6 mr/hour. At 14 ft above the surface of the pool (ceiling level) the estimated radiation level reduces to 0.6 mr/hour. These values are based on actual measurements made at the GULF GENERAL ATOMIC MARK F reactor at San Diego and include both direct radiation from the core and contributions from Ar-41 and N-16. The reactor tank water level safety system shall be used in conjunction with a reactor console display providing a digital read-out of the distance from the top of the reactor tank to the water surface, to satisfy this specification. A small rate of corrosion continuously occurs in a water-metal system. In order to limit this rate, and thereby extend the longevity and integrity of the fuel cladding, a water cleanup system is required. Experience with water quality control at many reactor facilities has shown that maintenance within the specified limits provides acceptable control. Additionally, by limiting the concentrations of dissolved materials in the water, the radiation field in the reactor room due to neutron activation of the dissolved materials is limited. This is consistent with the ALARA (As Low As Reasonably Achievable) philosophy. Operability for tank water level SCRAM is preformed prior to each run and monthly, Water quality is monitored prior to each run and monthly. Tank water temperature channels are checked semiannually and calibrated if required. Detailed electronic descriptions of the water monitoring systems can be found in Appendix D.1

7.1.4 Conclusions

The UUTR meets or exceeds the requirements for safe operation as required by the NRC through observance of the aforementioned water conditions that are verified as operational at start-up.

7.2 Power Monitoring Channels

The power monitoring channels include: Linear Power Channel Operation with Scram Function, Percent Power Channel Operation with Scram Function, Log Power Channel Function, Fission Chamber with Interlock Function, and Period Meter Function. The power monitoring channels interact with the control systems through the control rod drives, interlocks and limit switches for the drives and rods. All meters are located on the main control console in the control room.

7.2.1 Design Criteria

The redundant power monitoring channels provide the operator with an accurate power level reading during start-up, shut down and steady state operations. The interlocks and scram systems ensure the reactor level does not exceed the licensed full power 250 kW. The interlock system ensures the reactor is not brought to power without knowledge of the neutron population at very low levels. Technical Specification 3.1 The reactor shall not deliberately be raised above 250 kW under any conditions. Redundant power monitoring channels with SCRAM capability are checked prior to each run and calibrated semiannually to ensure operational and accurate power indicators. Technical Specifications 3.3.2, 3.3.3 state the reactor shall not be operated unless the two power monitoring/ safety channels are operable. The reactor shall not be operated unless the source count control interlocks are operable.

7.2.2 System Description

The UUTR is equipped with a compensated ion chamber and 2 uncompensated ion chambers that provide data to the linear power, percent power and log power channels respectively. The linear and percent power channels have digital readouts on the console in addition to the recorded graphical output. The integrated power channel uses the linear power output data to determine run integrated power. A fission chamber is used to determine the neutron start-up rate. If the level is less than 2 cps the interlocks restrict movement of the control rods.

7.2.2.1 Linear and Integrated Power

The linear power channel can display reactor power for 0.1 W to 1 MW. The numeric value displayed on the linear power channel is interpreted as a percentage of the power setting on the range switch. If the linear power exceeds 100% of the range switch setting a SCRAM is initiated. The output of the linear power channels' signal is integrated and displayed so the operator can monitor the run integrated power. A SCRAM will also occur if there is a loss of ion chamber supply voltage.

A compensated ion chamber is used with a transistorized linear amplifier and preamplifier to feed linear power circuits. Linear count rates are read on the linear recorder during reactor start-up, steady state power and shutdown. The compensated ion chamber (Westinghouse WL-6971 or equivalent) feeds a transistorized power-level scram amplifier. The compensated ion chamber has a thermal neutron sensitivity of 4.0 x 10^{-14} amp/n/cm^2·s and an uncompensated gamma sensitivity of 5.0 x 10^{-11} amp/R/hr.

The sensitivity of linear power channel is variable, so that its output on either the linear recorder or the meter will read 3, 10, 30, 100, 300, 1000, 3000, 10,000, 30,000, 100,000, or 300,000, 1,000,000, 3,000,000 W full scale, depending on the setting of the range change switch on the control panel.

The power level amplifier, fed by the linear power channel, is a Burr-Brown model 3061/25 integrated circuit operational amplifier with an accuracy of ± 0.02%, and a drift of 0.03% of full scale for 8 hrs within the temperature limits of 20 C to 50 C.

The integrated power channel uses the output of the linear power channel in conjunction with information from the range switch to exactly determine the reactor power level. This power level is integrated over time and fed to two digital counters, one of which can be reset by the operator to monitor run integrated power.

7.2.2.2 Percent Power

The percent power channel provides the reactor operator with reactor power information on a scale of percent licensed power. This channel is capable of scramming the reactor at a set point at or below 120% of licensed power. The percent power channel uses an uncompensated ion chamber (Westinghouse WL-6937 or equivalent) that feeds a transistorized power-level scram amplifier. The uncompensated ion chamber has a thermal neutron sensitivity of 4.0×10^{-14} amp/n/cm^2·s and a gamma sensitivity of 5.0×10^{-11} amp/R/hr. The percent power channel has a fixed sensitivity and its output is read on a meter indicating 0 to 150% of full power. This channel feeds a meter and the scram circuit directly through an attenuator.

The magnet scram amplifier is a General Atomic transistorized model AS-120. The scram is adjustable from 25% to 150% of full power. The accuracy of the trip point is ±5% of the set point with a maximum delay of 20 ms. The amplifier will handle up to 4-rod magnets. The scram adjustment on percent power channel will be set to scram at about 120% of full scale of the range switch. Scrams are also initiated for loss of ion chamber supply voltage.

7.2.2.3 Log Power and Period Meter

The log power channel provides the reactor operator with reactor power on a log scale. This meter provides the reactor operator with an additional indication of how quickly the reactors power is changing. The logarithmic channel (Log-N) provides continuous indication of power covering 5 decades from 4×10^{-4} to 4×10^{-9} amp. The Log-N uses an uncompensated ion chamber located approximately on the mid plane of the core on the outside of the perimeter core shroud. The detector is a Westinghouse WL-6337 or equivalent, having a thermal neutron sensitivity of 4.0×10^{-14} amp/n/cm^2·s and a gamma sensitivity of 5.0×10^{-11} amp/R/hr.

The signal for the period monitoring channel is provided from the period amplifier of the Log-N channel. The period amplifier is a General Atomic transistorized model AP-130 having a range of -40 seconds to +7 seconds. The period meter is located on the control console in the Control Room.

7.2.2.4 Fission Counter

The fission counter measures the number of neutrons in the core at start-up. An interlock is provided so that withdrawal of any control rods is not permitted, unless the source count is above the required minimum value of at least 2 counts per second. The fission counter shows the number of thermal neutrons that interact with the fission chamber every second. When a thermal neutron causes the U-235 lining of the fission chamber to fission, one of the fission products is propelled into the center of the chamber. The negative charge from the resulting ionization is collected on the center conductor. This pulse is amplified, discriminated from other ionization pulses (because it is much larger), and fed into a rate meter. If too few fission pulses are received, the source interlock is turned on.

The count rate channel provides information on the reactor at low power levels of about 10^{-3} W (1 count per second) to 2 W (10^4 counts per second). The detector is saturated at high power levels and is not used as reference past about 2 W. The count rate channel uses a fission counter, Westinghouse WL-6971 or equivalent, as the detector. This detector has a sensitivity of about 0.14 count/neutron/cm^2. The detector is approximately positioned on the mid plane of the core on the outside of the perimeter shroud of the core.

Additional count rate channel equipment includes a preamplifier (transistorized model), a linear amplifier (General Atomic transistorized model), a log count rate meter (General Atomic transistorized model), and a linear count rate meter (General Atomic model CR-100). Detailed electronic descriptions of the power monitoring channels can be found in Appendix D.2

7.2.3 System Performance Analysis

7.2.3.1 Linear and Integrated Power

The linear power level scram is provided as redundant protection against abnormally high fuel temperature and to assure that reactor operation stays within the licensed limits. The scrams at 100 percent of full operational power of 250 kW assures that the reactor operation will terminate well below a power level above which safe cooling may not be available. The integrated power provides an important element of the fuel's irradiation history that is used to determining the burn-up. The linear power SCRAM function is tested for operability prior to each reactor run. The power monitoring channels are checked semiannually and calibrated as required.

7.2.3.2 Percent Power

The percent power level scram is provided as redundant protection against abnormally high fuel temperature and to assure that reactor operation stays within the licensed limits. The percent power channel scrams at 120 percent of full operational power of 250 kW assures that the reactor operation will terminate well below that power level above which safe cooling may not be available and also at a level below the level of the temperature scram trip. The percent power SCRAM function is tested for operability prior to each reactor run. The power monitoring channels are checked semiannually and calibrated as required.

7.2.3.3 Log Power and Period Meter

The log power level is provided as redundant information/ protection against abnormally high fuel temperature. The log power channel is not a scrammable channel, but is checked for operability prior to each reactor run. The power monitoring channels are checked semiannually and calibrated as required. The period meter provides the operator with information as to period of the power level changes. The period meter is not a scrammable channel but is checked for operability prior to each reactor run.

7.2.3.4 Fission Counter

The fission counter insures that the reactor cannot enter an unstable operational mode if neutron count rate is too low to provide meaningful startup information. The fission counter is not a scrammable channel but is checked for operability prior to each reactor run.

7.2.4 Conclusions

Through the use of the two scrammable power channels and a source interlock, all of which must be verified as operational at startup, the UUTR meets or exceeds the requirements for safe operation as required by the UUTR Technical specifications and all applicable NRC Regulations.

7.3 Control Rod System

In this section the normal operating, scram, and interlock functions of the control rods will be discussed. The UUTR has three boron carbide control rods that are positioned by standard TRIGA electrically powered rack and pinion drives. The position of the each control rod is displayed on the console as a percentage of rod withdrawn. The control rods are held in place by electromagnetic when a scram is initiated the current is cut and the control rods drop by gravity into the core, shutting the reactor down.

7.3.1 Design Criteria

The control rods are designed to safely change reactor power and/or shut the reactor down. Technical Specifications 3.2 (2), 3.3.1, 3.3.2 , The maximum rate of reactivity insertion by control rod motion shall not exceed $0.30 per second. The scram time from the instant that a safety system setting is exceeded to the instant that the slowest scrammable control rod reaches its fully inserted position shall not exceed 2 seconds. The SCRAM time specification shall be considered to be satisfied when the sum of the response time of the slowest responding safety channel, plus the fall time of the slowest scrammable control rod, is less than or equal to 2 seconds. The reactor shall not be operated unless the startup count rate interlock, and control element withdrawal interlocks are operable. The startup count rate interlock was described in Section 7.2. The interlock exists to prevent the withdrawal of a control rod without some minimum count rate in the reactor. Control element withdrawal interlocks prevent the withdrawal of more than one control rod at once.

7.3.2 System Description

The control rods are positioned in the core by electromechanical devices that are controlled at the main console. A spring-loaded push button switch is used to raise the control rod and another is used to lower it. Each control rod has its own raise and lower buttons. The rod drive mechanism is an electric motor actuated linear drive equipped with a magnetic coupler. Each control rod has an individual scram button, additionally the controls rods can be scrammed as a group using the scram bar.

7.3.2.1 Control Rods

The UUTR utilizes boron carbide control rods characteristic of most TRIGA reactors. The rods are aluminum tubes approximately 43 inches long and are 1.35, 1.35, and .25 inches in diameter (safety, shim, and regulator rods respectively) with a powder boron carbide neutron absorber running the length of the rods. One rod is designated as a regulating rod and used for fine control during operation.

The control rods pass through normal fuel positions in the UUTR core in the top and bottom grid plates. Guide tubes ensure that the control rods remain in proper position during use.

7.3.2.1 Control Rod Drive Assemblies

The control rods are positioned by standard TRIGA electrically powered rack and pinion drives Figure 7.4.2.1. All rods and rod drives are identical and operate at a nominal rate of approximately 24 inches per minute. Limit switches mounted on each drive assembly stop the rod drive motor at the top and bottom of travel and provide switching for console indication which shows:

1. When the magnet is in the up position.
2. When the magnet (and thus the control rod) is in the down position.
3. When the control rod is in the down position.

A key-locked switch on the reactor console power supply prevents unauthorized operation of all control rod drives.

The rod drives are connected to the control rods through a connecting rod assembly. These assemblies contain a bolted connection at each end to accept the control rod at one end and the control rod drive at the other. The grid plates provide guidance for all control rods during operation of the reactor. No control rods can be inserted or removed by their drives in such a way that the rod would be disengaged from the grid plate.

Each drive consists of a stepping motor, a magnet rod-coupler, a rack and pinion gear system, and a ten-turn potentiometer used to provide an indication of rod position. The pinion gear engages a rack attached to a draw-tube which supports an electromagnet. The magnet engages a chrome-plated armature attached above the water level to the end of a connecting rod that fits into the connecting tube. The connecting tube extends down to the control rod. The magnet, its draw-tube, the armature, and the upper portion of the connecting rod are housed in a tubular barrel. The barrel extends below the control rod drive mounting plate with the lower end of the barrel serving as a mechanical stop to limit the downward travel of the control rod drive assembly. The lower section of the barrel contains an air snubber to dampen the shock of the scrammed rod. In the snubber section, the control rods are decelerated through a length of 3 in. The control rod can be withdrawn from the reactor core when the electromagnet is energized. When the reactor is scrammed, the electromagnet is de-energized and the armature is released.

The rod drive motors are stepping motors driven by a translator. The speed of the rods is adjustable and rods are normally set to insert or withdraw the control rods at a nominal rate 24 in./min. The unique characteristics of a stepping motor/translator system are used to provide fast stops and to limit coasting or over-travel.

A 110 v, 60 cps, two-phase motor drives a pinion gear and a 10 turn potentiometer. The potentiometer may be employed to provide rod position indications. The pinion engages a rack attached to the magnet draw tube. An electromagnet, mounted on the lower end of the draw tube, engages an iron armature that screws into the end of a long connecting rod which terminates at its lower end in the control rod. The magnet, armature, and upper portion of the connecting rod are housed in a tubular barrel that extends well below the reactor water line. Located part way down the connecting rod is a piston equipped with a stainless steel piston ring. Whereas the upper portion of the barrel is well ventilated to allow free movement of the piston in the water, the lower 2 in. of the barrel has graded vent ports to restrict piston velocity.

Clockwise (as viewed from the shaft end of the motor) rotation of the motor shaft rotates the pinion, thus raising the magnet draw tube. If the magnet is energized, the armature and connecting rod will rise with the draw tube, so that the control rod is withdrawn from the reactor core. The piston moves up with the connecting rod. In the event of a reactor scram, the magnet will be de-energized and will release the armature. The connecting rod, piston, and control rod will then drop, thus reinserting the control rod into the reactor. Since the upper portion of the barrel is well ventilated, the piston will move freely through this range. However, when the connecting rod is within 2 in. of the bottom of its travel, the piston is restrained by the dash pot action of the restricted ports in the lower end of the barrel. This restraint cushions bottoming impact.

7.3.2.2 Limit Switch

A spring-loaded pull rod extends vertically through a housing and up through the block. This rod terminates at its lower end in an adjustable foot that protrudes through a window in the side of the barrel. The foot is placed so as to be depressed by the armature when the connecting rod is fully lowered. Raising the rod releases the foot, allowing the pull rod to be driven upward by the force of the compression spring. The top of the pull rod terminates in a fixture that engages the actuating lever on a microswitch. As a result, the microswitch reverses position according to whether or not the armature (and control rod) is at its bottom limit. This microswitch is the rod down switch.

A push rod extends down through the block into the upper portion of the barrel. It is arranged so as to engage the top surface of the magnet assembly when the magnet draw tube is raised to its uppermost position. The upper end of the push rod is fitted with an adjustment screw that engages the actuator of a second microswitch. Thus, this microswitch reverses position according to whether the magnet is at or below its full up position. This microswitch is the magnet up switch.

A bracket, fitted with an adjustment screw, is mounted on top of the magnet draw tube. A third microswitch is arranged so that its actuating lever is operated by the adjustment screw on the bracket. The switch will thus reverse position according to whether the magnet draw tube is at or above its completely depressed position. This microswitch is the magnet down switch.

7.3.2.3 Circuit Operation

The circuit associated with the three micro switches provides Limit contacts for the motor and the control-rod enunciator system, which consists of three indicator lamps for each rod drive.

During normal operation, two points receive line power through the normally closed control rod UP and DOWN pushbuttons, which provide the dynamic braking. Depressing the UP button opens the line at a single point. This permits the line current to flow through the DOWN button to the second point through a 1 μf phase-shifting capacitor. The phase difference at the motor windings causes the motor to rotate in a clockwise direction. Counter clockwise motion is obtained when the control rod DOWN button is depressed.

The unconventional circuit employed in the rod-drive system minimizes the number of switch contacts required. Therefore, relays, with their attendant reliability problems, are not required. It should be noted that all rod-drive units are identical both mechanically and electrically: they are, therefore, interchangeable.

The motor coupling is attached to the motor shaft by a single 8-32 dog-point setscrew. Both the pinion gear and the potentiometer coupling are pinned to the motor coupling. To prevent the follower potentiometer from supporting any of the pinion-gear load, the potentiometer coupling runs in an outrigger bearing. The follower-potentiometer shaft is connected to the potentiometer coupling by a single 6-32 setscrew. An oil-saturated, felt vapor seal restricts the entrance of water vapor into the follower potentiometer. Gravity loading of the rack against the pinion ensures minimum backlash between the rack and the follower potentiometer. Both motor and follower potentiometer are fully enclosed in metal.

7.3.2.4 Position Indicators

LED readouts are provided to indicate the position of each of the control rods in the core. The readout indicates, by percentage, how far the control rod is out of the core. The push button switches are also lit to indicate the status of the control rod. The lower button (red) is lit when the control rod is fully inserted into the core. The raise button is lit when the control rod is in any position other than fully inserted.

7.3.2.5 Interlock Systems

The push buttons for the control rods are interlocked in such a way that all three rods can be lowered at the same time in order to shut down the reactor quickly without a scram, yet only one of the control rods can be raised at a time due to an electric interlock between the three raise switches. This prevents excessive reactivity from being inserted into the reactor in a short amount of time.

The minimum source-count interlock relay prevents the withdrawal of all rods. The source interlock operates in the following manner. Part of the line is connected to the UP pushbuttons of each rod drive through a source relay. If minimum source count is reached, the relay is de-energized and the interlock makes the switches inoperative.

7.3.3 System Performance Analysis

TS 3.2(1) Ensures that power increases caused by rod motion will be terminated by the reactor safety system before the fuel temperature safety limit is exceeded. The worst-case reactivity insertion accident from unrestrained motion of the control rods would be related to reactor startup, between 1 and 100 mW, with a subcritical condition corresponding to the source level. In such a case, 30¢/sec of reactivity insertion continues until the power level scram is tripped. Following a further delay of 0.1 seconds, the control rods begin to insert reactivity at the rate of $3/sec until the rods are fully inserted. Assuming no thermodynamic feedback occurs (making the calculation quite conservative) 1.7 mW-sec of energy is produced by the excursion, raising the fuel temperature by 20°F in the peak fuel flux location of the core. This temperature rise is far below the allowable rise to the fuel damage point starting from ambient conditions at startup.

Control rod motion is limited by the drive system to 24 in/min. The interlock between upward movement of more than control rod simultaneously also ensure that the reactivity insertion rate is not exceeded.

TS 3.3.1 SCRAM time less than 2 seconds ensures that the reactor will be promptly shut down when a scram signal is initiated. Operation experience and analysis have indicated that for the range of transients anticipated for a TRIGA rector, the specified scram time is adequate to ensure the safety of the reactor. SCRAM times are checked semiannually.

The start-up channel interlock with the control rods ensure shutdown if neutron count rate is too low to provide meaningful startup information.

7.3.4 Conclusions

These rod drives were first developed in 1959, and have been modified and improved a number of times. The design has proven to be reliable and has been used in more than 60 TRIGA reactors containing over 160 rod drives. The UUTR control rod system meets or exceeds the requirements for safe operation as required by the NRC

7.4 Fuel Temperature Channels

The UUTR is equipped with two independent instrumented fuel elements that monitor the fuel temperature in the core. The temperature of the fuel is displayed on the reactor console. Exceeding the set point will initiate a SCRAM. Set points for fuel temperature are set will below the limited safety system settings.

7.4.1 Design Criteria

Technical Specifications 2.1 defines the safety limits. The peak fuel temperature in a stainless-steel clad, high hydride fuel element shall not exceed 1000 °C under any conditions of operation. The peak fuel temperature in an aluminum clad low hydride fuel element shall not exceed 530 °C under any conditions of operation. Technical Specification 2.2 defines the limited safety system settings to ensure the safety limits are never reached. For a core composed entirely of stainless steel clad, high hydride fuel elements or a core composed of stainless steel clad, high hydride fuel elements with low hydride fuel elements in the F or G hexagonal ring only, limiting safety system settings apply according to the location of the instrumented fuel as indicated in Table 7.4.1. For a core containing flux traps, the limiting safety system settings given in either of the above two tables must be applied to the anticipated hottest fuel element in the core. In addition, TS 3.2.2 and 3.3.3 requires measuring one fuel temperature for operation with SCRAM settings at or below the limited safety system setting.

Table 7.4.1
Locations and Limited Safety Settings for Stainless Steel Clad and Aluminum Clad Fuel

Location of Instrumented Fuel Rod	Limiting Safety System Setting for stainless steel clad	Limited safety system setting for aluminum clad
B-hexagonal ring	800 °C	460 °C
C-hexagonal ring	755 °C	435 °C
D-hexagonal ring	680 °C	390 °C
E-hexagonal ring	580 °C	340 °C

7.4.2 System Description

The fuel temperature monitoring channels consists of a K type thermocouple and an Omega CN9000A temperature controller. The useful range is 0 °C to 800°C with a ±1°C accuracy. The controllers are configured for on/off control. The output relays of the two fuel temperature controllers are connected in series in a normally energized closed circuit to provide the fuel temperature SCRAM function. Fuel temperatures are calibrated annually.

If the thermocouple calibration procedure indicates that any temperature channel is incorrect by more than one or two degrees, the deviation can be corrected by adjusting the offset or gain constants internal to the controller. Please refer to the CN9000A manual for this procedure. Handling of the thermocouple wires and connectors may result thermal gradients, which will disturb the channel readings slightly. Temperatures must be given time to stabilize (several minutes) after any handling of the wiring before calibration may proceed. Fuel temperatures channels are checked during each run and compared to previous runs of the same power level as early check for drift or temperature channel malfunction. Fuel temperature setpoints for the SCRAM function are set at 200 °C for 100 kW operation. Setpoints for the SCRAM functions at 250 °C kW will be set a 300 °C.

7.4.3 System Performance Analysis

The most important parameter for a TRIGA reactor is the fuel rod temperature. A loss in the integrity of the fuel rod cladding may arise occur if there is a buildup of excessive pressure between the fuel moderator and the cladding, and the fuel temperature then exceeds the safety limit. Such pressure is caused by the presence of air, fission product gases, and hydrogen from the dissociation of the hydrogen and zirconium in the fuel moderator. The magnitude of this pressure is determined by the fuel-moderator temperature and the ratio of hydrogen to zirconium in the alloy.

The safety limit for the high hydride TRIGA fuel is based primarily on experimental evidence obtained during high performance reactor tests on this fuel. These data indicates that the stress in the cladding due to hydrogen pressure from the disassociation of zirconium hydride will remain below the stress limit, provided that the temperature of the fuel does not exceed 1150° C and the fuel cladding is water cooled. See GA-9064, Safety Analysis Report for the Torrey Pines TRIGA Mark III Reactor, submitted under Docket No. 50-227 for more detailed information.

The safety limit for the low hydride fuel elements depends upon avoiding the phase change in the zirconium hydride which might cause excessive distortion of a fuel element. This phase change takes place at 500° C as shown by the phase diagram on page 5-13 of the University of Utah SAR 1985. Additional information is given in "Technical Foundations of the TRIGA" Report GA-471, pages 63-72, August 1958. After reviewing all safety analysis conducted for the preparation of Chapter 13 of this report (Accident Analysis), we conclude that no credible accident can cause melting in the fuel's cladding.

For Stainless steel clad, high hydride fuel element, the limiting safety system settings that are indicated represent values of the temperature which, if exceeded, shall cause the reactor safety system to initiate a reactor scram. Since the fuel element temperature is measured by fuel elements designed for this purpose, the limiting settings are given for different locations in the fuel array. Under these conditions, it is assumed that the core is loaded such that the maximum fuel temperature is produced in the B-hexagonal ring.

The margin between the safety limit of 1000°C and the limiting safety system setting of 800°C in the B-hexagonal ring was selected to ensure that conditions would not arise which would allow the fuel element temperature to approach the safety limit. The safety margin of 200°C accounts for differences between the measured peak temperature and calculated peak temperature encountered during operation and for uncertainty in temperature channel calibration. The thermocouples that measure the fuel-moderator temperature are located approximately midway between the fuel axial center line and the fuel edge.

During steady-state operations, the equilibrium temperature is determined by the power level, the physical dimensions and properties of the fuel element, and the parameters of the coolant. Because of the interrelationship of the fuel-moderator temperature, the power level, the changes in reactivity required to increase or maintain a given power level, any unwarranted increase in the power level would result in a relatively slow increase in the fuel-moderator temperature. The margin between the maximum setting and safety limit would assure a shutdown before conditions could result that might damage the fuel elements.

For low hydride fuel element, the 460°C maximum limit for the safety system setting gives an ample margin that assures that the safety limit would not be reached through errors in measurement. Temperatures of 460°C have been shown to be safe through extensive operating experience. The surveillance requirement on the measurements of the fuel dimensions will give control over changes that result from the thermal cycling during operation. The temperature shown for C, D and E hexagonal ring locations were derived using the power distributions from report GA-4339.

However, experimental data based upon 363 critical reactor operations indicate a maximum fuel temperature of 114°C at 100 kW and an extrapolated maximum fuel temperature of 314°C at 300 kW. Additionally the operation experience over the reactor lifetime gives assurance that the thermocouple measurements of fuel element temperature have been reliable.

7.4.4 Conclusions

The UUTR meets or exceeds the requirements for safe operation as required by the NRC by its utilization of the two scrammable fuel temperatures, both of which are verified as operational at startup. The fuel temperatures are also checked during each reactor run and compared to a previous reactor run at the same power level.

7.5 Reactor Console

The UUTR reactor console has a variety of safety functions that can initiate scrams manually or automatically in addition to previously discussed. These functions are: Manual, and Key Scrams, Loss of Console Power Scram, Loss of High Voltage Scram.

7.5.1 Design Criteria

Both the manual and automatic scram functions of the reactor console are designed to provide safe shut down of the reactor under any credible situation. The manual option allows the operator to initiate a scram at their own discretion. Likewise, should the console itself malfunction an automatic scram will be initiated, resulting in loss of console power and thus, of magnet power.

Technical Specification 3.3.3 states the reactor shall not be operated unless the safety system channels (Manual, and Key Scrams, Loss of Console Power Scram, Loss of High Voltage Scram) are operable. A channel check of each of the above reactor safety system channels shall be performed before each day's operation or before each operation extending more than 1 day, except for the pool level channel which shall be tested monthly or at intervals not to exceed six weeks. All reactor safety channels shall undergo a channel test and a channel check after any maintenance or modification.

7.5.2 System Description

The reactor SCRAMs are normally closed circuits that, when interrupted, cause the reactor control rods to be dropped back into the core thereby shutting the reactor down. In general, each SCRAM is a normally closed relay contact that is connected from terminal strip #1 position X to terminal strip #2 position X. The relay for the linear channel SCRAM, for example, is located in the linear channel display controller. All SCRAM logic, including TS-1 and TS-2, is executed at 120 VAC for historical reasons (the relays used have 120 VAC coils). Disconnect console power to service. Each SCRAM has its own relay that is set up to latch itself on with the current flowing through the normally closed SCRAM relay. Once a SCRAM has occurred, a contact of this relay serves to turn on the appropriate SCRAM indicator on the center console. The scrams can be reset by momentarily turning the Reset key to the Reset position. When the console power is first turned on all of the SCRAM lights will be activated so that the operator can check that none of the bulbs have burned out. All relays used in the SCRAM logic are normally energized so that a local loss of power or loose connection will cause a SCRAM rather than possibly preventing one.

High Voltage Power Supply:

The high voltage power supply is the only remaining piece of the original General Atomics designed TRIGA console. When the high voltage fails (or the High Voltage SCRAM Test button is pressed) this relay will open and cause a high voltage SCRAM.

Data Acquisition System:

A Fluke 2620 multichannel voltmeter is connected to all of the temperature, power and rod drive indicator channels in the console. Physically the unit resides in the back of the console above the circuit bin. The data acquisition system provides an easy method for maintenance personnel to examine critical signals in the console, and it can also be connected to a computer via an IEEE-488 or RS232 link to record and display reactor operations data. The unit has 20 channels (plus a channel zero at its front panel banana plugs) that can be configured for VDC, VAC, resistance, frequency, current, and thermocouple measurements. For more information on the operation and programming of the 2620, see the manual in the equipment filing cabinet. Some of its maintenance uses are:

- To double check the temperature monitor displays, using the thermocouple function.
- To check for a short in a thermocouple (particularly an instrumented fuel element) using the resistance function.
- To check for excessive noise on any of the power, temperature, or rod position lines using the VAC function.
- For general trouble shooting in the console by plugging the test leads into the front panel, and using channel zero.

7.5.3 System Performance Analysis

Manual scram allows the operator to shut down the system if an unsafe or abnormal condition occurs. In the event of failure of the console power supply, the console power supply scram provides that operation will not continue without adequate instrumentation. The channel tests will ensure that the safety system channels are operable on a daily basis or for an extended run. Channel tests and checks after maintenance and modification assure the operability and reliability of the channel.

7.5.4 Conclusions

The UUTR meets or exceeds the requirements for safe operation as required by the NRC by utilization of the aforementioned manual and automatic safety systems.

7.6 Engineered Safety Features Actuation Systems

The ventilation system is an engineered safety system. A description of the system is found in Chapter 6, Engineered Safety Features. The ventilation system operates in two conditions: normal and alarm. Normally, the ventilation system provides a negative pressure difference between the reactor room and the surroundings. Under alarm conditions, dampers close off the inlet air that enters the facility, thus increasing the negative pressure difference. The alarm condition is activated when area radiation levels exceed setpoints at any of the area radiation detectors. Activation of the alarm condition when the reactor is not in operation involves the following: an alarm bell will be activated in the reactor room and in the area directly above the reactor room, the ventilation damper will close, [redacted]. During reactor operations, the alarm signal is routed to the control console where the reactor operator is notified.

7.6.1 Design Criteria

The negative pressure assures that any radioactive gases produced in the reactor are not released to the environment without first being monitored. Gases will not leave the facility through any unmonitored cracks. All gases will leave the facility through the ventilation duct monitored by the continuous air monitor system. This monitoring device has alarms with conservative setpoints designed to allow reactor operators to take corrective actions to prevent exceeding regulatory release limits. The ventilation fan motor can be manually operated from the control console and is required to be operating during reactor operations.

Technical Specification 3.5 , 4.3.4 and 5.6 the reactor shall not be operated unless the facility ventilation system is in operation, except for periods of time not to exceed 48 hours to permit repair or testing of the ventilation system. In the event of a substantial release of airborne radioactivity within the facility, the ventilation system will be secured or operated in the dilution mode to prevent the release of a significant quantity of airborne radioactivity from the facility.

7.6.2 System Description

Normal operation of the TRIGA reactor involves the production of radioactive gases. The two gases of concern are Argon-41 and Nitrogen-16. Nitrogen-16 is produced when oxygen-16 captures a neutron and releases a positron (n,β+). Due to the depth of the TRIGA pool and the short half life of N-16 (7.1 sec), the radiological hazard to persons both in and outside of the facility is negligible.

Ar-41 production occurs in a similar manner. Due to the solubility of air in water, the reactor pool has trace amounts of Argon (i.e. Ar-40). This undergoes neutron capture, creating Ar-41. This isotope has a 1.83 hour half-life. Therefore, it can easily be detected as it migrates to the surface. In order for the TRIGA reactor to operate in a safe manner, the Argon-41 must be discharged from the facility. This is the main task of the facility's ventilation system. It removes the Ar-41 and thus reduces the exposure of CENTER personnel to Ar-41.

The CENTER ventilation system operates in two different modes; normal and emergency. In each of these modes, the goal is to keep the controlled access area (CAA) in a state of negative pressure with respect to the atmosphere and the Merrill Engineering Building.

In normal operating condition (the time designated by the operating of the TRIGA reactor), the air in the controlled access area (CAA) is constantly being exchanged. The air leaving the facility has a volumetric flow rate of 1385.0 cubic ft. per minute. The result of this is a negative pressure of greater than 0.01 inches of water in the CAA.

The CENTER ventilation system draws its supply air from the Merrill Engineering Building's main ventilation air, Prior to entry in the controlled access area, building air passes through one of two pre-filters (position 2). This reduces the concentration of dust in the air so that the pool water contamination can be kept at a minimum. Then, air mixes with the current air in the CAA and migrates to point 3, where it passes through one of two pre-filters and then HEPA filters (points 4 and 5) At this point, an air sample is diverted from the ventilation system to the continuous air monitor (point 6). Here, airborne activity is determined and relayed to the TRIGA control console. The same data is also stored permanently on a strip chart recorder located on the CAM system.

Waste air from the CAM is pumped back into the ventilation system and mixed with air from the reactor room. This air, after leaving the reactor room, the duct enters the 1205 K directly north of the reactor room. The duct then rises vertically to the building room where the fans and motors are located. The duct is vented approximately 40 feet above ground level where it is expelled 10 feet above the penthouse on the roof (point 7) and dispersed into the atmosphere. The ventilation system draws air from the CAA through the fumehood. This allows chemical processing of experiments to take place at the same time as TRIGA reactor irradiations. The fumehoods in 1205 are equipped with a differential pressure gauge. Air pressure is sampled on each side of the HEPA. When the pressure differs by 2.0 inches of water, it is time to change the HEPA filters. Used HEPA filters are surveyed.

The only process which may result in the production of radioactive gases is the operation of the TRIGA reactor. Hence, there may be times in which the blower on the roof is OFF. In this mode the air in the CAA has a reduced volumetric flow rate of 346.4 cubic feet per minute. In the event of elevated radiation levels, the system will automatically be switched into emergency mode.

In the event that airborne activity in the CENTER CAA exceeds the preset level of 10 mR/hr, the ventilation system will switch to emergency mode operation. This provides an enhanced negative pressure to the CAA of greater than 0.1 inches of water. This is to ensure that all potential airborne radionuclides are contained in the facility and the exhaust air then goes through the HEPA filters. From there, the air is monitored by the CAM that records the radiation levels of any release of radioactive material.

After an evaluation is made by the responding SRO, the system may be returned to normal operating conditions. This is accomplished by depressing the damper reset button located on the TRIGA control console.

7.6.3 System Performance Analysis

During normal operation of the ventilation system the concentration of Argon-41 in unrestricted areas is below Derived Air concentrations and maximum effluent (Maximum Permissible Concentration). In the event of a substantial release of fission products, the ventilation system will be secured automatically. Therefore, operation of the reactor with the ventilation system shutdown for short periods of time to make repairs insures the same degree of control of release of radioactive materials. Regular testing of confinement equipment and the regular replacement of all ventilation filters will assure that the confinement of radioactive releases can be attained if needed.

7.6.4 Conclusions

Use of the ventilation system adds additional protection to the staff and public during both routine and emergency operating conditions. This insures that the UUTR meets or exceeds the requirements for safe operation as required by the NRC.

7.7 Radiation Monitoring Systems

The UUTR has two important radiation detection systems to ensure that the reactor will operate within established guidelines. These systems are the Area Radiation Monitors System, and the Continuous Air Monitor System. The Area Radiation Monitors are scrammable channels.

7.7.1 Design Criteria

Readouts from two different radiation monitoring systems are a part of the reactor console. The Area Radiation Monitors (ARMs) display the radiation levels present at four strategic areas in the CENTER. Should the limiting safety system setting of 10mR/hr be exceeded, the high radiation alarm will be triggered. The Continuous Air Monitor (CAM) draws air from the facility ventilation system and tests it for radioactive noble gas, radioactive Iodine, and radioactive airborne particulates. Any alarm will sound at the console and at the CAM if the Limiting Safety System Settings for this unit are exceeded.

Technical Specification 3.3.2: The reactor shall not be operated unless the following radiation monitoring systems are available and operable: A Continuous Air Monitor (CAM) consisting of the following detectors: Noble Gas detector, Particulate detector, and Fission Product detector. The following Area Radiation Monitors (ARMs) which shall be operating and have both readouts and alarms on the reactor console: Reactor Tank Top, Reactor Room Ceiling above Reactor Tank, Reactor Room Exhaust Duct, Counting Laboratory. The Area Radiation Monitoring equipment (ARMs) and the Continuous Air Monitoring system (CAM) shall be calibrated biennially and shall be verified to be operable at monthly intervals, or at intervals not to exceed six weeks.

7.7.2 System Description

Area Radiation Monitor:

The Area Radiation Monitor (ARM) is located on the TRIGA control console. Remote detectors are placed in areas of the CENTER where personnel will be working with radioactive material. These locations are: the reactor ceiling, reactor tank, the stack, and the counting lab. In the event that radiation levels exceed 10.0 mR/Hr at any location, the high radiation alarm will be activated. This sends a signal to the security system that will implement the following:

- SCRAMS the reactor (if in operation)
- [redacted]
- Closes the inlet air damper
- Activates audible and visual High Radiation alarms

Closing the inlet damper for the CAA is done automatically by sending a 12 VDC pulse from the ADT system to the damper motor control box. The 12 VDC pulse energizes L-1 (24VAC). L-1 closes and latches LR-1 (latching relay 1). This results in the de-energizing of the damper motor, thus closing the CENTER facility ventilation. This resets LR-1 and closes the circuit to provide power to the damper motor, thus opening the damper. Once the alarm has been implemented, the enunciator can be secured by using the damper bypass key (this only stops the audible and visual alarm). In the event of main power failure, a battery backup system takes over for a limited amount of time. For extended periods of time, power is taken from the back up generator located on a concrete pad outside of the CENTER lab.

Continuous Air Monitor:

The Continuous Air Monitor is located in the radiochemistry lab. For a complete description of its function, see the manual in the equipment filing cabinet. A cable consisting of multiple twisted pairs connects the CAM with the displays on the console left front panel. Inside each of the three CAM modules, an op amp drives a current setting potentiometer that is in series with the respective current sensitive meter on the console. By holding the particular CAM module in the calibrate mode one can verify that the reading of the meter at the console agrees with that on the CAM. Calibrating the CAM should not change the agreement of the CAM meter and the console meter. Adjust the current setting potentiometer in the CAM if, for some reason the two do not agree. The CAM alarm relays have been configured to turn on the red light DS22 and sound an alarm BZ2 (both 120 VAC) at the console if any of the three CAM monitors exceeds its setpoint. An alarm and light will also be activated at the CAM itself. The green light DS23 is normally on and will only turn off if one of the CAM channels goes below its lower setpoint (usually indicative of a defective detector).

Equipment specifications are:

Area Radiation Monitors:

- Useful Range: 0.01 to 100 mR/hr
- Sensor Type: Geiger-Mueller Tube
- Accuracy: ±20% for gamma energies from 40 keV to 2.5 MeV
- Calibration Interval: Annual
- Safety Functions: Exceeding 10 mR/hr causes SCRAM (Channel can be bypassed)

Continuous Air Monitors:

- Useful Range: 1 to 10^5 CPM
- Sensor Type: Geiger Mueller Tube except for Iodine channel which is a NaI crystal
- Accuracy: ±10%
- Calibration Interval: Annual

7.7.3 System Performance Analysis

When the Continuous Air Monitor is operating, it samples the reactor room exhaust air prior to the HEPA filter located in the ventilation system, providing information on the levels of Ar-41, particulates, and Iodine in the reactor facility. The ARMs provide a continuous evaluation of the radiation levels in the reactor facility and provide warning alarms when the radiation levels exceed anticipated levels. The ARM located in the Counting Laboratory provides a continuous evaluation of the radiation level at that location and provides warning alarms when the radiation levels exceed anticipated levels. Experience has shown that monthly verification of area radiation and air-monitoring setpoints in conjunction with annual calibration is adequate to correct for any variation in the system caused by a change of operating characteristics over a long time span.

7.7.4 Conclusions

Using two radiation monitoring systems both of which must be verified as operational at startup, the UUTR meets or exceeds the requirements for safe operation as required by the NRC.

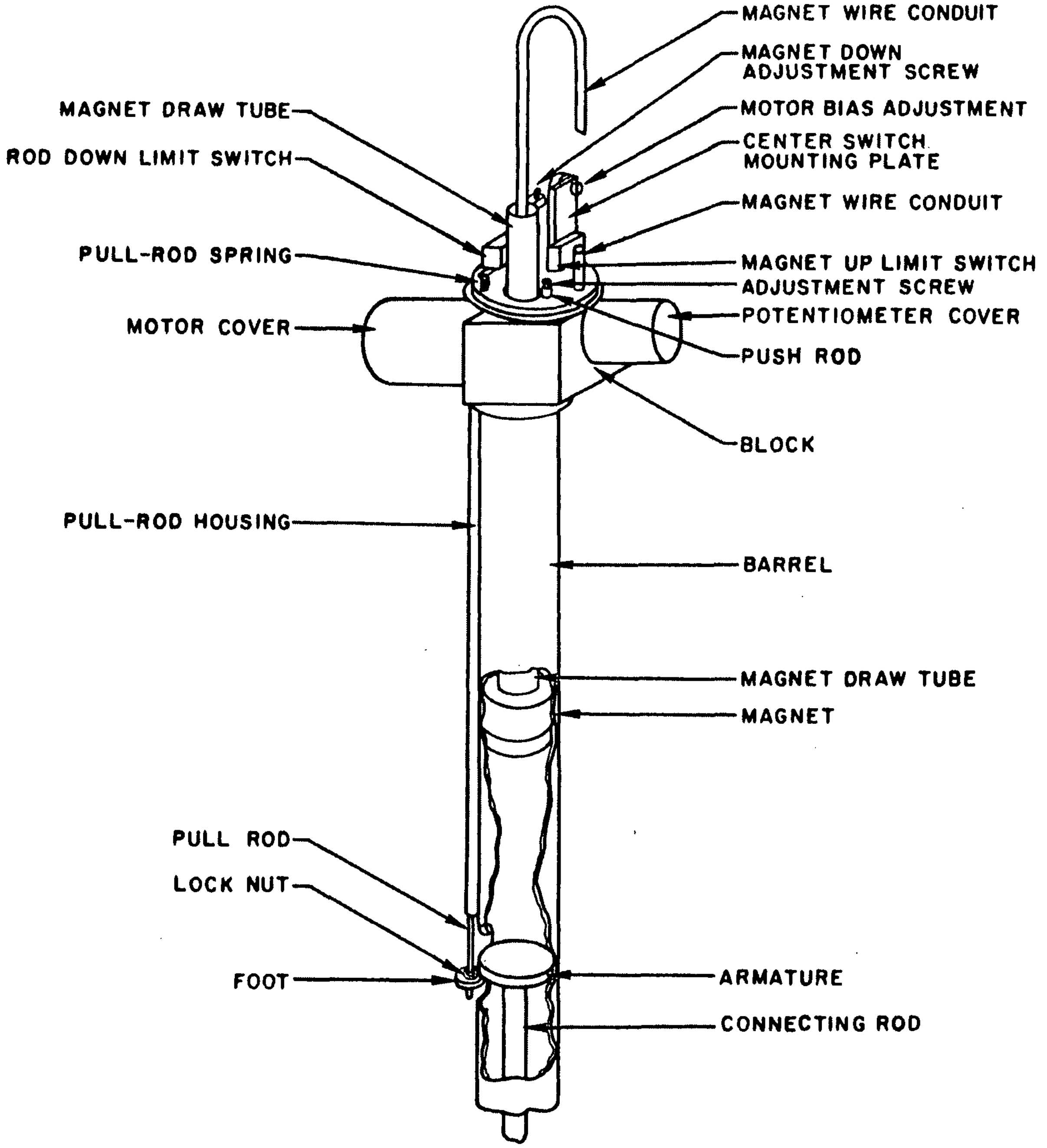

Figure 7.5.2.

Control Rod Drive

Chapter 8

Electrical Power

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8 ELECTRICAL POWER

8.1 Normal Electrical Power Systems

The electrical power for the UUTR is supplied from the Merrill Engineering Building's electrical power system. The electrical power provided for building lighting and reactor instrumentation is single-phase, 60 Hz, 120/240 V. The reactor room has its own independent circuit panel which is controlled and monitored by reactor personnel.

The design and safety equipment of the UUTR does not require building electrical power to safely shut down the reactor, nor does the UUTR require building electrical power to maintain acceptable shutdown conditions.

8.2 Emergency Electrical Power Systems

The reactor will scram in the case of an building electrical power interruption. The emergency power is not required to maintain the reactor in a safe shutdown condition. The radioactive decay heat generated even after extended runs in the core following a scram is not sufficient to cause fuel damage. Power for the radiation monitors and the facility intrusion detectors is supplied by an uninterruptible power supply (UPS) within the reactor room. In the event of an electrical outage, this UPS supplies the necessary power for the operation of these instruments for a minimum of 24 hours. Battery-powered emergency lighting is also available to facilitate personnel movement during a power outage.

Chapter 9

Auxiliary Systems

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9 Auxiliary Systems

Several auxiliary systems aid in the safe operation of the reactor: lighting, cooling, heating, electricity and other services. This chapter will discuss the features of these systems and their use.

9.1 Heating, Ventilation, and Air Conditioning Systems

University plant operations provide the reactor lab with electricity, potable water, heating, and air-conditioning. University maintenance personnel maintain these services. For a description of the facility ventilation system see Chapter Six, on Engineered Safety Systems.

9.2 Handling and Storage of Reactor Fuel

This section of the SAR discusses the fuel handling, the fuel handling tools, the fuel protection design features, criticality safety, fuel security, fuel inspection criteria.

9.2.1 Fuel Handling Cycle

The fuel handling system provides a safe and effective means for transporting and handling the reactor fuel from the time it enters the boundaries of the UUTR facility until it leaves. The fuel handling cycle within the UUTR consists of (1) receiving fresh, unirradiated, fuel elements, (2) transferring the fresh fuel elements into the reactor in-tank storage racks or storage pits by use of the fuel element handling tool, (3) unloading used fuel elements from the reactor grid into the in-tank storage racks, (4) loading the fresh fuel elements from the in-tank storage racks into the reactor grid, (5) repositioning fuel elements within the reactor grid, (6) interchanging fuel elements between the reactor grid and the in-tank storage racks, (7) transferring irradiated fuel elements from the reactor in-tank storage rack by use of the fuel transfer cask and the overhead handling system to the fuel storage pits in the floor of the reactor room, and (8) transferring fuel from either the storage pits or in-tank storage racks to a shipping cask for removal. This section presents the safety aspects of those handling operations.

Fresh, unirradiated fuel arrives at the UUTR facility in Department of Transportation approved shipping containers. The fresh fuel, is removed from the shipping containers by hand and stored until needed.

All handling of fuel within the reactor tank is accomplished by use of the fuel element handling tool (FEHT), with the exception of the instrumented fuel elements. The reactor room overhead crane and the fuel transfer cask are used to transfer irradiated fuel elements between the in-tank storage racks and the fuel storage pits.

The reactor room overhead crane is used to position the fuel transfer cask in the reactor tank such that the cask top is approximately 9 ft below the water level. With the lead plug removed, the FEHT is used to lift an irradiated fuel element from an in-tank fuel storage rack and place it into the cask. After replacing the transfer cask lead plug, the reactor room crane is used to raise the cask out of the tank and transport it to a position over a fuel storage pit. Using the FEHT, the fuel element is raised from the bottom door of the transfer cask, allowing the bottom door to be opened. The fuel element is lowered out of the cask into its storage location. The reverse operation is used to remove an irradiated element from the storage pits and place it in the fuel transfer cask. Appropriate radiation monitoring will be conducted during this operation in order to assure that doses are kept as low as reasonably achievable.

An approved shipping cask will be used to transport irradiated fuel elements from the UUTR as needed. Much of the same above-described equipment is used to transfer an irradiated fuel element from the in-tank storage racks to the UUTR storage pits, and to place an irradiated fuel element in the shipping cask.

The first step in this operation is to load an irradiated fuel element into the fuel transfer cask, as described above, from the storage pits or in-tank storage racks. The fuel element transfer cask is then mated to the top of the shipping cask. This is the same procedure used for the transfer of a fuel element from the transfer cask to the storage pits. All fuel movement within the facility requires the use of procedures that have been previously reviewed and approved by the Reactor Safety Committee. These procedures are designed to insure the safe efficient movement of fuel and to prevent fuel damage and personnel exposure.

9.2.2 Fuel Handling Equipment

The fuel handling system provides a safe, effective, and reliable means of transporting and handling reactor fuel from the time it enters the UUTR facility until it leaves. To accomplish safe fuel movement UUTR has a vendor built FEHT, an overhead crane, a fuel handling cask, and fuel storage racks in the reactor tank.

9.2.1.1 Fuel Handling Tool

Tools are provided for handling individual fuel elements and for manipulating other core components. Individual fuel elements are handled with a flexible or rigid handling tool. The FEHT utilizes a locking ball-detent grapple to attach to the top end fitting of a fuel element.

9.2.1.2 Overhead Crane

An overhead crane running on tracks provides the capability for movement of heavy objects (including the handling of the fuel element cask) anywhere in the reactor room. The crane has a capacity of 4000 pounds and is locally controlled from a pendant box but can be computer controlled. The reactor room crane will be operated in accordance with ANSI B30.11, Monorail Systems and under hanging Cranes. In addition, any slings required to transfer the fuel cask will be used in accordance with 29 CFR Part 1910.184, Slings.

9.2.1.3 Fuel Transfer Cask

A shielded fuel transfer cask is used to transfer irradiated fuel elements from the reactor tank to the spent fuel storage pits or to a shipping cask. The fuel transfer cask is both top and bottom loading and holds either one fuel element or an instrumented fuel element. The structural components are fabricated from stainless steel with a lead filler. The maximum radiation exposure rate is about 5 mR/hr (gamma) at the outer surface of the transfer cask when it is loaded with an irradiated fuel element that has been allowed a six-month cooling time after operating in the highest flux region of the core for one year at one megawatt power. The internal components of the cask that contact the fuel are fabricated from stainless steel. Cask-lifting lugs have been designed using the ASME code for analysis guidelines. This analysis shows that the maximum shear load between the lifting lug and cask weld is less than 1000 lb//in^2 of weld area when the entire weight of the cask is on one lug. The allowable load for this weld is 6360 lb/in^2. of weld, a safety margin greater than 6 even with the conservative assumption that all weight is on one lug. The cask lugs have been load tested in accordance with NE F8-6T.

9.2.2 Fuel Storage

9.2.2.1 Fuel Storage Racks

The in-tank fuel storage for the UUTR consists of six, in-tank, aluminum fuel storage racks, with a combined capacity to accommodate 78 irradiated fuel elements. The in-tank fuel storage racks are located at the outer edge of the reactor tank. These storage racks are designed to meet the following criteria:

(a) The in-tank fuel storage racks are designed with sufficient spacing between fuel elements to ensure that the array, when fully loaded, will be substantially subcritical. (Storage requirements are KEFF < 0.8)

(b) The in-tank fuel storage racks are designed to withstand earthquake loading to prevent damage and minimize distortion of the rack arrangement.

(c) The in-tank fuel storage racks have a combined capacity for storage of a typical core loading of irradiated fuel elements.

(d) The in-tank fuel storage racks are mounted on the inside of the reactor tank and are deep enough below the water surface to provide adequate radiation shielding.

(e) The in-tank fuel storage racks are designed and arranged to permit efficient handling of fuel elements during insertion, removal, or interchange of fuel elements.

The fuel elements are loaded into the in-tank fuel storage racks from above. Each storage hole has adequate clearance for inserting or withdrawing a fuel element without interference. The weight of the fuel elements is supported by the lower plates of the racks. Each in-tank fuel storage rack is securely hung from a fixture in the reactor tank by two hanger type attachments. These fixtures are securely welded to the interior tank wall. This mounting arrangement prevents the racks from tipping or being laterally displaced.

Within a fuel storage rack, control of spacing is not actually required to limit the effective multiplication factor of the array (Keff). The in-tank fuel storage racks are configured such that criticality is not possible. Furthermore, 2 racks of 8.5 w% fuel stored back to back are subcritical (i.e., K <= 0.74 for twice the U-235 mass). In the unlikely event of loss of reactor tank coolant water, the loss of the water moderator would increase the safety margin by reducing the Keff. The in-tank fuel storage racks are made of polyethylene and are designed to withstand a UBC Zone 3 earthquake, when fully loaded.

9.2.2.2 Fuel Storage Pits

Additional fuel storage at UUTR is maintained in three fuel storage pits, [redacted] The storage pits are sufficiently spaced and shielded with hydrogenous material to insure that there is no possibility of neutron coupling between the storage pits.. Each pit has a liner and a lead-filled shield plug that will be locked in place when fuel is not being moved into or out of the pits. The pits have racks with holes for holding fuel elements. Each hole in the rack can only hold one fuel element. All storage pit material (liners, racks, plug casing, and pipes) that may contact either the fuel elements or the pit water are fabricated from aluminum or 304 stainless steel. This is the same type of material as used for the fuel element cladding and end fittings. The fuel storage pits were designed with the following criteria.

- The spent fuel storage pits are designed with sufficient spacing to ensure that the stored fuel array, EVEN when fully loaded, will be independent of each other and subcritical Keff < 0.8.

- The spent fuel storage pits are designed to withstand earthquake loading to prevent
- damage and distortion of the pit arrangement.

- [redacted]

- The spent fuel storage pits are fabricated from materials compatible with the fuel elements and shall provide adequate personnel shielding.

- The spent fuel storage pits are designed and arranged to permit efficient handling of fuel elements during insertion or removal of fuel elements.

- The spent fuel storage pits have shield plugs that can be locked in place.

The fuel elements are loaded into the racks from above. Each hole in the rack has adequate clearance for inserting or withdrawing a fuel element without interference. The lower plates of the racks that are supported by the pit liners support the weight of the fuel elements. Each rack is designed so that it is constrained by the pit liner and cannot tip or become laterally displaced.

Analysis shows that the [redacted] fuel elements, which have been in the core operating at 100 kW, can be removed from the reactor tank after one day of decay and safely stored in a single pit either with or without water (dry). Analysis shows a significant decrease in fuel temperature by allowing the fuel to decay 10 days after shutdown prior to being moved to storage. Therefore, allowing at least 10 days of decay prior to transferring fuel to the storage pits will provide a margin of safety so that external radiation doses from the fuel in the transfer cask and the fuel's decay heat temperature will remain within safe limits.

Within a fuel storage pit filled with water, control of spacing is not required to limit the effective multiplication factor of the array (Keff). An analysis shows the largest Keff for a pit is approximately 0.75 when all five pits are loaded to capacity with 8.5 wt % fuel elements [redacted] and are full of water (0.45 when dry). Since 20/20 fuel contains erbium, it is similar in reactivity to 8.5 wt %; thus, there should be no significant changes to the criticality of the storage pits. Furthermore, [redacted] elements is only 1/3 the number required for criticality. Radiation levels at the reactor room floor level with either water in the storage pits or the lead plug in place are below 2 mR/hr.

The spent fuel storage pits are designed to withstand horizontal and vertical accelerations due to earthquakes. Stresses in a fully loaded storage pit will not exceed stresses specified by the UBC Zone 3 seismic criteria.

9.2.3 Fuel Inspection

Fuel inspections of all UUTR fuel elements are made on a biannual basis. These inspections are necessary to insure that the integrity of the fuel cladding is maintained throughout the fuel life at the UUTR facility. These inspections look for defects and surface anomalies of the fuel elements' cladding. Based upon changes observed, the fuel will be left in its current location or removed from the core. If a fuel element is suspected of being damaged but the extent is unclear, the fuel will be considered damaged if any of the following three criteria is met. These criteria are:

(a) If the transverse bend, its sagitta exceeds 0.125 inches over the length of the cladding.

(b) In measuring the elongation, its length exceeds its original length by 0.25 inches.

(c) A clad defect exists as indicated by release of fission products.

A jig exists in the facility to measure criteria a and b, and monthly water quality inspections exist to measure criteria c.

9.2.4 Fuel Security

To prevent fuel theft, adequate design, operation, and administrative controls have been established at the UUTR facility.

9.3 Fire Protection Systems and Programs

The design basis for the UUTR fire protection system is to provide a detection and suppression capability which will mitigate any losses should a fire develop. It should be noted that fire protection is not required to accomplish a safe shutdown of the reactor or to maintain a safe shutdown condition.

Both detection and suppression systems installed in accordance with National Fire Protection Code are utilized in the UUTR. The fire protection system consists of smoke detectors and sprinklers, which are located throughout the facility. Whenever one of the fire detection devices activates, visual and audible warning devices alarm throughout the facility. The fire detection and suppression system is maintained by University Plant operations.

Additional fire response is provided by the Salt Lake City Fire Department. The Salt Lake City Fire Department are instructed periodically on the special needs of fire protection at the UUTR facility, by the UUTR staff.

By the design nature of a pool type reactor and the material construction of the fuel storage pits, the release of radiological material from the facility is severely limited. All radioactive source material is stored in locked fire proof cabinets to limit the possibility of release in the event of a fire.

9.4 Communications

Communications within the laboratory are provided by the University telephone system. There are four separate lines that enable the control room to telephone the office of the Reactor Supervisor and the offices of the Senior Reactor Operators. The telephone sets are also equipped with an intercom function.

9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material

The UUTR staff has materials used to maintain and operate the TRIGA reactor in compliance with the facilities operating license No. R-126 pursuant to 10 CFR parts 30, 50 and 70. This facility is licensed to receive, possess, and use up to [redacted] kilograms of U-235 and a [redacted] sealed Pu-Be source (A request has been made to increase the U-235 holdings to [redacted] kg). Additionally, the facility is licensed to receive and possess, but not separate, such byproduct and special nuclear material as may be produced by operation of the reactor.

Samples and sources are stored in shielding materials that reduce the activity levels at the surface of the container. Storage containers range from small lead "pigs" to large brick vaults. These containers may be located in the reactor room, the radiochemistry labs, or radiation measurement lab. Radioactive samples and sources are clearly labeled and secured. Access to rooms containing radioactive sources is restricted to trained personnel and those accompanied by trained personnel. Access to the reactor room and control room is restricted to licensed operators and those accompanied by licensed operators.

Samples, experimental devices, reactor components, or by products of normal reactor operations can only be released to holders of a radioactive materials license. Irradiated samples must undergo a materials release survey. The container type, dose rates, license identification, and other appropriate information must be noted on form CENTER-027.

The UUTR facilities include four compensating fume hoods that operate with a minimum air velocity of 100 cfm per fume hood. Each fume hood can be retrofitted with the appropriate shielding as required. Additionally, each lab is equipped with separate waste collectors for radioactive and hazardous waste. Administrative controls prohibit processing or mixing of waste that creates "mixed waste". Hazardous chemicals are characterized and disposed of through the University of Utah's Environmental Health and Safety Department. Radioactive waste is characterized and disposed of through the University of Utah's Radiological Health Department.

9.6 Cover Gas Control in Closed Primary Coolant Systems

The UUTR is an open pool type reactor with the primary coolant loop open to the atmosphere. As a result, no cover gas control system is necessary for this reactor.

9.7 Other Auxiliary Systems

The UUTR facility has several auxiliary systems that do not fall into standard format categories these are: the facility lighting, access control. These topics will be individually addressed below.

9.7.1 Lighting Systems

Fluorescent lighting is used in all of the rooms associated with reactor operations. University maintenance personnel service this lighting. High intensity lights are located at the top of the reactor to illuminate the core for visual inspections and various other operations including movement of experiments, movement of the source, and maintenance of the core and its associated fixtures.

There are three, battery-powered, emergency lighting units located in the reactor facility to provide lighting in the event of power failure. University maintenance personnel also service the emergency lighting. (The maintenance personnel must be accompanied for such activities in the CENTER).

9.7.2 Access Control

The reactor room is equipped with an security system in accordance with the requirements of the CENTER security plan. [redacted] Persons entering the reactor room are required to wear personal dosimeters or be accompanied by an individual wearing dosimetry. Emergency exit may be made through the radiochemistry laboratory or through the control room.

Chapter 10

Experimental Facilities And Utilization

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10 EXPERIMENTAL FACILITIES AND UTILIZATION

10.1 Summary Description

The primary purpose of the UUTR facility's experimental program is to promote education and research. To accomplish these goals, the UUTR facility is equipped with a central irradiator, three beam ports, a pneumatic tube irradiator, a fast irradiator, and a dry tube irradiator. To date, no beam tube irradiations have been performed at UUTR. Monitoring of experimental activities can be preformed from the reactor console location using reactor control instrumentation, visual observation, voice, intercom, and remote camera. All experiments that the UUTR performs fall into one of three categories: routine, modified routine, and new. All experiments began as new experiments. New experiments must be reviewed and approved by the Reactor Safety Committee prior to implementation. Modified routine experiments are similar in nature to experiments that are routinely performed. They require the review of a Senior Reactor Operator prior to implementation. Routine experiments are those experiments that have been approved and documented and are performed on a regular basis.

10.2 Experimental Facilities

The experimental facilities listed below are given into three categories: in-core irradiators, in-reflector irradiators, and external irradiators. The Central Irradiation Facility (CIF), Pneumatic Transfer System, and vacant fuel positions are all in-core irradiators. The Dry Tube Thermal Irradiator (TI) and the Fast Irradiator (FI) are in-reflector facilities. The beam ports are UUTR's external facility.

10.2.1 In-Core Irradiators

The UUTR has been designed with multiple in-core irradiation facilities to facilitate a broad range of potential experimental activities. These facilities consist of a central cavity, the pneumatic transfer tube, and individual fuel element locations. Each of these facilities are individually described below.

10.2.1.1 Central Irradiation Facility (CIF)

The central irradiation facility (CIF) is located in the central fuel pin position. A special tube has been constructed to accommodate samples and can be placed in the central fuel pin position (the A fuel ring) by means of a cable. The dimensions of this assembly are the same as a fuel pin. Because this facility is an in-core irradiator located in the center of the core, there are special restrictions on the reactivity of samples placed in this irradiator.

Additionally, a cutout is available where both the A and B rings are removed to accommodate a larger irradiator, capable of holding multiple samples. This irradiator has two special features associated with it. One of these is a sealed interior that holds heavy water. The other is a motor that rotates the sample holder to spatially average the neutron fluence in the assembly. This cutout facility is currently not in the UUTR core and will require an appropriate analysis, experimental review and approval by the reactor safety committee prior to implementation.

10.2.1.2 Pneumatic Transfer System (PTS)

A pneumatic transfer system (PTS) is available for use at the UUTR facility. The PTS is installed within a 1.5 inch O. D. tube and is driven by the force of dry, compressed helium. The PTS has a slight curve in its tube in order to prevent direct streaming of neutrons from the core to the surface of the pool.

The UUTR PTS is designed to quickly transfer individual specimens into and out of the reactor core. The specimens are placed in a small polyethylene holder, "rabbit," which in turn is placed into the receiver. The rabbit travels through aluminum and PVC tubing to the terminus at reactor core centerline, and returns along the same path to the receiver. Directional gas flow moves the rabbit between receiver and terminus. A compressed gas system supplies helium to the system, and a set solenoid valve directs flow. Controls to operate the compressed gas and solenoid valve are on the console. The key system elements and their functions are described below.

The "rabbit" is an enclosed polyethylene holder. Experiments are inserted into the rabbit and contained by a screw cap on one end. Available space inside the rabbit is approximately 0.625 in. in diameter and 4.5 in. in length.

The receiver positions the rabbit for transfer to the terminus and receives the rabbit after irradiation. Two transfer lines connect the receiver to the terminus: one allows the rabbit to travel between the receiver and terminus, the other controls gas flow direction.

The receiver is located in the counting laboratory. The exhaust is released into the ventilation stack and prevents uncontrolled release of airborne radioactivity. The exhaust fans maintain the negative pressure with respect to the surrounding room. The PTS exhaust passes through a pre-filter, a HEPA filter, before continuing up the stack. The stack monitor and the CAM sample the exhaust air in the ventilation system including the exhaust released from the PTS.

The terminus consists of two concentric tubes, which extend into the reactor core. The inner tube is perforated with holes (which are smaller than the sample container diameter). The bottom of the inner tube contains a stainless steel spring shock absorber to lessen the impact of the rabbit when it reaches this end of the transfer line, which is approximately at the mid-plane of the core. When air flows to the terminus, the capsule rests in the bottom of the inner tube: when air flows to the receiver, the capsule moves out of the inner tube by air flowing through the tube's holes. The outer tube supports the inner tube and provides a path for the air to flow through.

The outer tube bottom support is shaped like the bottom of a fuel element and can fit into any fuel location in the core lattice. Both tubes that extend to the top of the reactor tank are offset to reduce radiation streaming. A weight has been installed to counteract the buoyancy of the air-filled tubes and keep the terminus firmly positioned in the core. A set of solenoid valves direct flow through the transfer-line-loop sending the rabbit either to the terminus or to the receiver depending on valve position.

10.2.1.3 Vacant Fuel Positions

Reactor grid positions that are vacant of fuel elements may be utilized for the irradiation of materials. These in-core irradiation facilities or the positioning of a single experiment in a fuel element vacancy grid position shall meet the requirements of the Technical Specifications for design, safety evaluation, restrictions and approvals.

10.2.2 In-Reflector Irradiators

The UUTR has been designed with multiple in-reflector irradiation facilities to facilitate a broad range of potential experimental activities. These facilities consist of a dry tube thermal irradiator and a fast irradiator.

10.2.2.1 Dry Tube Thermal Irradiator

A dry tube thermal irradiator is available for use in the trapezoidal shaped D_2O tank attached to the side of the core. The samples are placed into polyethylene vials attached to a line and dropped into the irradiator through a curved PVC tube that extends to the top of the reactor pool.

10.2.2.2 Fast Neutron Irradiator

The fast neutron irradiator is designed to provide sample exposure to neutrons with minimal moderation. The entire device has two pieces: a stand and a sample holder. All structures were fabricated from Al-5052, which is a material compatible with the TRIGA reactor system.

The irradiator was constructed in two pieces: the outer box and the inner box. The outer box is loaded with lead bricks and sealed by bolting the inner box to the outer box with aluminum bolts. Graphite gasket material ("Grafoil") was used on the contact surfaces. Grafoil is radiation resistant and has been proven in high radiation applications at other reactor facilities.

The irradiator is loaded with standard lead shielding bricks on one side. This side is placed next to the core face for additional gamma shielding. Also, thermal neutron absorbers may be placed in any position to decrease sample exposure to thermal neutrons. The irradiator is about 500 pounds sub-buoyant.

The sample holder contains one lead brick above the test volume and one lead brick below the test volume. The amount of shielding may be adjusted as necessary. The sample holder was sealed with neoprene gasket material. It is about 35 pounds sub-buoyant when loaded with the two lead bricks inside.

The device stand rests on the floor of the reactor tank, and the irradiator is located on the device stand. The assembled device aligns with the reactor core the western most hexagonal face. All aluminum surfaces were anodized to prevent corrosion. The stand and the irradiator will remain in the pool during their useful lifetime and may be moved to change the irradiation position or may be temporarily removed from the pool for maintenance of the radiation conditioning materials. Only the sample holder is moved on a regular basis to access samples.

The irradiator is located on the outside of the grid structure. The irradiator affects the flux by changing the moderation and reflectivity in the region where the irradiator replaces light water with air, lead, and aluminum. As installed the irradiator reduces core reactivity by no more than -$0.10. The device is considered to be a secured experiment. The sample holder is also considered to be secured and is estimated to change the reactivity by no more than -$0.05 when placed in position. These reactivities cannot result in a prompt critical condition if the devices are accidentally separated from the core or flooded by water.

The irradiator, located outside the grid structure, does not interfere with fuel cooling channels; therefore, it does not affect fuel and cladding temperatures or fuel internal pressure. Use of the fast neutron irradiator does not increase the probability of personnel exposure because it introduces no additional mechanisms for exposure, nor does it increase the probability of radioactive material release because it introduces no additional mechanisms for release. Use of the fast neutron irradiator does not increase the probability of pool water leakage because it does not introduce any new mechanisms for failure of the tank.

Because of the low reactivity worth of the sample holder, the holder may be conservatively removed while the reactor is either critical or shutdown. However, the sample holder shall be inserted and removed from the irradiator only while the reactor is shutdown in order to carefully control neutron exposure.

Samples to be irradiated in the sample holder will be in compliance with TS 3.6 and will have less than $0.95 of reactivity. The hazards associated with the fast neutron irradiator device have been reviewed by the UUTR staff and it was concluded that the installation of the fast neutron irradiator did not constitute a change in the Technical Specifications and has no unreviewed safety issues.

10.2.3 External Irradiators

The UUTR facility has only one experimental facility for irradiation that is external to the tank and biological shielding supplied by the tank water, the three beam ports. The CENTER has no immediate plans to open any of these three beam ports. Future use sill require analysis, documentation, review and approval of utilization of the beam ports by the reactor safety committee.

10.2.3.1 Beam Ports

The reactor system contains three diagonally directed beam ports that extend from the reactor core to the reactor floor. The upper beam port is adequately shielded with sand and capped at the reactor floor level with a 1/8 inch thick steel cap for security. When the beam port is employed, the sand will be removed and a sealed aluminum beam tube (carefully weighted so as to have a net density greater than water) will be installed between the inner tank wall and the reactor core shroud along the common axis of the upper beam port. There are three aluminum beam tubes (one for each port) currently in storage. Each beam tube is composed of two sections aligned along a common axis. The top tube section is a 1 foot diameter tube that will be inserted in the port between the reactor floor and the wall of the aluminum reactor tank. If installed, this tube would not penetrate the aluminum tank because it is sealed at the end where it makes contact with the tank. During beam port use, both the lower and upper beam tube sections can be sealed against air flow to ensure that no radiation hazards arising from argon-41 or nitrogen-16 buildup will be present. Furthermore, penetration of the reactor tank is not necessary to install and utilize the beam tubes. Only inner tubes have contact with the reactor tank, and thus there is no increased risk of inner tank water leakage. When the beam port is employed, appropriate shielding at the reactor floor level will be constructed with dense concrete blocks and other available shielding materials.

10.3 Experiment Review

Prior to conducting any new irradiation experiment, a technical and safety review of the proposed experiment must be performed. The following procedures apply to the review and approval of all experiments utilizing any of the UUTR irradiation facilities. These procedures outline the prerequisites to experimental approval, the review and approval of new, modified routine and routine experiments. Additionally, the procedures that outline documentation, and personnel training requirements are presented in the section below.

10.3.1 Prerequisites to Experiment Approval

No new experiment shall be implemented until the following criteria are met:

1) A hazards analysis of the proposed experiment has been performed and the results have been reviewed for compliance with the limitations on experiments (TS 3.6) by the Reactor Safety Committee (RSC).

2) An experiment review has been completed by the operating staff and has received RS approval. Minor modifications to a reviewed and approved experiment may be made at the discretion of the SRO. If the SRO determines that the changes do not constitute a significantly new or different safety risk greater than the approved original experiment, then the modified experiment may be conducted without further approval. The SRO must document all such decisions (TS 6.8).

3) The reactivity worth of a proposed experiment has been calculated and does not exceed Technical Specifications 3.6 (1). If the reactivity worth of an experiment has been measured in a similar core position at equal neutron flux, then an estimate of the reactivity worth is acceptable.

4) The quantity of known explosives is less than 25 milligrams and the pressure produced in the experiment container upon accidental detonation of the explosive has been experimentally determined to be less than the design pressure of the container.

10.3.2 Review and Approval of New, Modified Routine, and Routine Experiments

The approval procedure applied to an experiment or class of experiments shall depend on whether a proposed experiment is evaluated as a new, modified routine, or routine experiment.

The review and approval of all experiments must be initiated by the experimenter who shall complete and submit the Form CENTER-027, Irradiation Request and Performance, to the RS. The RS shall verify the completeness of the requested information and pass the form to Operating Staff for evaluation. The Operating Staff shall evaluate the form and shall determine if the experiment requires initial RSC approval in the form of an approved Experiment Authorization or if the experiment may can be reviewed and approved by CENTER Staff under an existing experimental authorization.

10.3.2.1 New Experiment

A new experiment is any proposed activity utilizing the UUTR that does not directly conform to an existing Experiment Authorization and therefore requires RSC approval. This approval is initiated by the submittal of a completed Experiment Authorization UUTR staff. The Experiment Authorization will then be evaluated by UUTR staff. The US NRC requires that new experiments that entail unreviewed safety questions must be approved by their organization as well. The UUTR staff shall then review the request at a staff meeting. Final approval comes from the RSC after reviewing the Experiment Authorization, TRIGA reactor EA form, and the minutes of the UUTR staff meeting. If approved, then the approved new experiment shall be passed to the RS and operating staff for scheduling. The experimenter must then submit a Form CENTER-027 and the new experiment shall receive final approval by the signature of the RO on this form.

10.3.2.2 Modified Routine Experiment

A modified routine experiment is one where planned or desired changes to the experiment could result in an increase of a safety hazard previously identified for the routine experiment. But the modified experiment remains within the parameters of an existing Experiment Authorization since the experiment presents no new and unreviewed safety hazards. These modifications can be approved by an SRO.

10.3.2.3 Routine Experiment

A routine experiment is one that has an existing approval from the RSC and has an existing Experiment Authorization and TRIGA Reactor experiment authorization form. To perform a routine experiment, the approval signature of a RO on Form CENTER-027 is required.

10.3.3 Documentation

This section describes the forms that are required for the documentation of experiment review and approval.

10.3.3.1 Experiment Authorization

An Experiment Authorization must be submitted for any new experiment. The authorization shall include a description of the experimental devices and general procedures that will be used to conduct the proposed experiment. It must also include an analysis of safety hazards. The authorization shall be submitted to CENTER staff for screening before it is submitted to the RSC for approval. This is a standard form for staff and RSC review of an experiment. The RS must sign this form. For new and modified experiments, additional authorization signatures are required from the RA and the RSO or their designees. This form is used as a training tool and a checklist to insure the experiment integrates with Technical Specifications, Experimental Authorization, and CENTER Procedures as well as other applicable CENTER documents. Completion of this form with all appropriate signatures shall give approval to CENTER operations to perform the proposed experiment.

10.3.3.2 Irradiation and Performance Request Form CENTER-027

The submittal CENTER form –027 is required in order to request irradiation or services of an approved experiment. The purpose of this form is to describe the isotopes to be produced, the activity expected, and information regarding the handling of radioactive materials. This form must be signed and approved by an RO.

The sample, its encapsulation, the procedures for handling it, and the method of positioning it in the reactor must satisfy the following criteria in order to be approved for irradiation. This criteria is established to provide the individuals reviewing Form CENTER-027 an adequate technical basis for making their decision.

1) Encapsulation must ensure sample containment in order to prevent contamination of the reactor pool, handling areas, and any laboratories involved.

2) The induced sample activity can be safely handled using available equipment.

3) If a dimensional change of the sample is expected, adequate expansion space must be left in the irradiation capsule.

4) The expected reactivity change due to insertion and removal must be within acceptable limits.

5) Significant reactivity variations due to sample movement (sway, bobbing, or rotation) must be prevented.

6) Expected activity calculation must be figured as either the total activity of all samples or a per batch amount from which the total activity can be calculated. The activity should be categorized according to the major contributing isotopes.

7) Review and approval documentation for the experiment must be on file in the control room.

Activity level should be calculated at the end of irradiation (i.e., decay time equals zero).

10.4 Personnel Training Requirements

The training and requalification of CENTER staff for conducting an experiment is documented in the RO Requalification Program. Non-operator personnel who are involved with an experiment while it is being conducted must be briefed prior to an experiment and closely supervised at all times by a qualified licensed reactor operator.

Chapter 11

Radiation Protection Program And Waste Management

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(Figures and Tables are numbered according to the sections that refer to them.)

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

This chapter deals with the UUTR radiation protection program and the corresponding program for management of radioactive waste. Specifically addressed are the radiation sources that will be present during normal operation and the expected radiation exposures due to normal operation. The chapter also describes the facility radiation protection programs used to monitor and control these sources and exposures. The majority of the detailed calculations supporting this chapter are contained in Appendix A, B, C.

11.1 Radiation Protection

The purpose of the UUTR radiation protection program is to allow the maximum beneficial use of radiation sources with minimum radiation exposure to personnel that are consistent with the UUTR ALARA program. Requirements and procedures set forth in this program are designed to meet the following fundamental principles of radiation protection:

- Justification - No activity potentially resulting in some radiation exposure shall be adopted unless the activity produces a net positive benefit.
- Optimization - All exposures shall be kept as low as reasonably achievable (ALARA).
- Limitation - The dose equivalent to individuals shall not exceed limits established by appropriate state and federal agencies. These limits shall include, but not be limited to those set forth in the Code of Federal Regulations (CFR).

The radiation protection measures used at the UUTR are patterned after other TRIGA reactor facilities where the radiation sources and activities are similar. Facility organization charts, actual radiation measurements and operating data from around the UUTR, and a description of radiation protection program components will be used to characterize the features of the different programs used to maintain occupational doses and releases of radioactivity to the unrestricted environment as low as reasonably achievable (ALARA). See the facility organization chart, Figure 12.1.1, in Chapter 12 of this report. For a more detailed description of the facilities organization see Chapters 12 and 14 (Conduct of Operations, and Technical Specifications).

11.2 Radiation Sources

The radiation sources present at the UUTR can be categorized as air-borne, liquid, or solid sources. Each of these categories will be discussed individually in sections 11.2.1 through 11.2.3, but the major contributors to each category can be summarized as follows: Airborne sources at the UUTR consist mainly of Argon-41 (Ar-41), due largely to neutron activation of air (contains about 1% argon) dissolved in the reactor's primary coolant, and Nitrogen-16 (N-16), due to neutron interactions with oxygen (O-16) in the primary coolant. Liquid sources are quite limited at the UUTR and include mainly mop water, some activation samples, and liquid scintillation samples. No routine liquid effluent or liquid waste is anticipated. Solid sources are more diverse, as they are associated with most TRIGA reactor operations. Such sources include the fuel in the 250 kW core, and fresh unirradiated fuel. In addition, other solid sources are present such as the neutron startup source, small fission chambers for use with nuclear instrumentation, items irradiated as part of normal reactor use, and small instrument check and calibration sources. Solid waste is yet another solid source, but is expected to be very limited in volume and curie content as past history at the UUTR as demonstrated. Solid waste is expected to result almost exclusively from activation.

11.2.1 Airborne Radiation Sources

During normal operation of the UUTR, there are two sources of airborne radioactivity, Ar-41 and N-16 that are generated. The assumptions and calculations used to assess the production and radiological impact of these air-borne sources during normal operations are detailed in Appendix B. Therefore, that information will only be summarized in this section.

Fuel element failure, although very unlikely, can occur while the reactor is operating normally, abnormally, or when shut down. Such a failure could usually result from a manufacturing defect or corrosion or damage of the cladding. The expected result would be a small penetration of the cladding through which fission products would be slowly released into the primary coolant. Some of these fission products, primarily the noble gases, could slowly diffuse through the cooling water into the air of the reactor room. Because of the large inventory of coolant water in the UUTR tank (~8000 gallons), migration of fission product gasses to the air is very limited and slow. Although this type of failure could occur during normal operation, its occurrence would be evident from the pool tank radiation monitors and reactor room air monitors. No further reactor operation would take place after such an event until the situation had been evaluated and eliminated (i.e., the failed element located and removed from the core). The failure of a single element, for any reason, is discussed and evaluated in Chapter 13 and Appendix C as an abnormal situation or an accident.

11.2.1.1 Argon and Nitrogen Production in Experimental Facilities

Production of Ar-41 and N-16 can occur in the reactor. Therefore, an evaluation of the production and release of these gases was conducted. Listed below are the maximum concentrations of waste gases allowed to be released into unrestricted and restricted areas as stipulated in 10 CFR 20.

Ar-41: 1×10^{-8} (μCi/ml) Unrestricted
Ar-41: 3×10^{-6} (μCi/ml) Restricted
N-16: *
N-16: *
* denotes no 10 CFR 20 limits established; hemispherical immersion model used.

The pneumatic system uses compressed helium for all cases that were explored and the reactor is considered to be operating at a maximum power of 250 kW. The steady state production of Ar-41 is 8.25×10^{-7} μCi/cm^3 and 1.08×10^{-7} μCi/cm^3 for N-16. This release of Ar-41 is with in regulations for restricted areas. The use of air to drive the pneumatic system also meets 10 CFR 20 requirements. The Ar-41 produced in the section of the pneumatic transfer system is exhausted from the system through a HEPA filter to the UUTR facility stack on the roof of Merrill Engineering Building. The stack is approximately 40 feet-ground level and insures rapid mixing and dispersal. There has been no significant increase in measurements attributable to Ar-41 releases, as measured by the stack monitor for the past two decades of operations of the UUTR. Therefore, the Ar-41 from the pneumatic transfer system is not considered to be a measurable contributor to unrestricted radiation doses associated with UUTR operations. Use of air for the pneumatic system meets 10 CFR 20 requirements. See Appendix B for calculations and assumptions.

Two other experimental facilities exist in which Ar-41 production can occur. These are the dry tube irradiator and the beam ports. Argon production in the dry tube irradiator is analogous to the case of the pneumatic transfer tube except that Ar-41 transport from the tube must occur by diffusion into the reactor room before entering the exhaust stacks. The beam ports remained capped and there are no current plans to open the existing beam ports. Before future use of these beam ports, review and approval for such an application must be obtained from the UUTR Reactor Safety Committee and the NRC if deemed necessary.

11.2.1.2 Argon Production from the Pool Water

Argon-41 in the reactor room occurs as the irradiated argon evolves from the primary coolant into the air of the room. This evolution results from the reduced solubility of argon in water as the water temperature increases. Detailed calculations addressing the production and evolution of Ar-41 from the primary coolant may be found in Appendix B.

At 250 kW, assuming complete mixing of the Ar-41 with reactor room air, the equilibrium Ar-41 concentration in the reactor room with the room exhaust system on, is approximately $1.11 \cdot x\ 10^{-8}$ μCi/ml.

The 10 CFR Part 20 Derived Air Concentration (DAC) for a semi-infinite cloud of Ar-41 is $3\ x \cdot 10^{-6}$ μCi/ml restricted and $1.0 \cdot x\ 10^{-8}$ μCi/ml unrestricted. Estimated dose based on conversion factors for submersion in a semi-infinite cloud of radioactive noble gases the dose rate is $2.44\ x10^{-12}$ Sv per hour ($2.44\ x10^{-14}$ REM/hour) (ICRP publication 30).

Actual measurements of Ar-41 in the reactor room after reactor operation for about 4.0 hours at 90 kW (reactor room exhaust system on) showed Ar-41 concentrations averaging about $2.67 \cdot 10^{-8}$ μCi/ml for areas that are occupied during normal work in the room. This would then correlate to about $2.77 \cdot x\ 10^{-8}$ μCi/ml at 250 kW, corresponding to $5.9 \cdot 10^{-18}$ Sv.

11.2.1.3 Ar-41 Release to the Unrestricted Area

The Ar-41 from the reactor room is discharged from the UUTR through the facility's exhaust stack, which is 40 feet above ground level. Dilution with other building ventilation air and atmospheric dilution will reduce the Ar-41 concentration considerably before the exhaust plume returns to ground level locations that could be occupied by personnel. The detailed calculations relating to the dispersion of Ar-41 released from the stack are contained in Appendix B.

Using the approach detailed in UUTR SAR 1985, the Ar-41 concentration in the reactor room air from the activation of Argon - 40 dissolved in the water for 250 kW operation is

($1.11\ x10^{-8}$ μCi/cm^3) a

And Ar-41 released through the ventilation system is

9.25×10^{-3} μCi/sec.

Estimated dose rate in the reactor room based on conversion factors for submersion in a semi-infinite cloud of radioactive noble gases is 2.44×10^{-12} Sv per hour (2.44×10^{-14} REM/hour) (ICRP publication 30).

The maximum release from experimental facilities of Ar-41 in the reactor room exhaust air was calculated to be at a concentration of ~ 0 μCi/ml. In addition, from B.2, a steady concentration from the pool water of 1.11×10^{-8} μCi/cm^3 was predicted. Thus the maximum total is 1.11×10^{-8} μCi/cm^3.

The Ar-41 concentration for this position assuming it coincides with the cloud centerline (i.e. assume conservatively that y = h = 0) and x = 350 meters is found to be $1.9 \cdot 10^{-9}$ μCi/ml, which is five times less than the unrestricted limits of $1.00 \cdot 10^{-8}$ μCi/ml. This conservative estimate does not account for "building dilution," dilution by other exhaust fans operating on the Merrill Engineering roof (estimated to provide further dilution by at least a factor of 10 at all times), radioactive decay of the Argon-41 during transport, or the fact that the reactor will be operated approximately 20 hours per month.

11.2.1.4 Nitrogen-16 Production from Pool Water

N^{16} is produced in the reactor tank water by two principle neutron capture reactions:

$$N^{15} + n^{l} \Rightarrow N^{16}$$
$$O^{16} + n^{l} \Rightarrow N^{16} + p^{l}$$

The calculation of N^{16} production and dose from the tank from these reactions requires several detailed calculations. N^{16} generation in the core was calculated first. Next, the coolant flow rate in the core (also given in Chapter 4 and Appendix A) was used to calculate the coolant transport time from the core to the surface. Combining the results of these calculations, a surface dose rate and a dose rate at a point two meters above the center of the tank were calculated from the N^{16} distributed in the tank water volume. Then, the transport of N^{16} from the pool to the air was calculated.

At a power level of 250 kW, the concentration of N^{16} in the core was estimated to be 1831 atoms or an activity of 4.6×10^{-9} Ci/sec. The number of N-16 atoms leaving the pool in 7.6×10^{-9} atoms/(sec cm^3) and the dose is 7.1×10^{-9} mRem/sec.

11.2.2 Liquid Radioactive Sources

Under normal operating conditions, there is no liquid released from the reactor pool or the cooling loop. Dissolved minerals and metals are removed in the resin beds, characterized and transferred to Radiological Health for disposal. Mop water from reactor room floor cleaning is collected in a sub grade holding tank, characterized and released or transferred based on the activity of the water. Spent liquid samples are also characterized and transferred to Radiological Health for disposal. The requested increase in power level will not substantially change the level and amount of liquid radioactive sources produced for disposal.

11.2.3 Solid Radioactive Sources

The solid radioactive sources associated with the UUTR program are summarized in include low enriched fuel, start source, calibration sources used for instrumentation and samples. Solid radioactive sources are secured and inventoried.

11.2.4 Exposure During Normal Operations

Operation of the University of Utah TRIGA Reactor will create a source of radiation in the form of particles emitted from the core. In all conceivable cases, gamma ray emission and neutron radiation are the primary components of radiation fields external to the core. Beta and alpha radiation will be completely absorbed by the materials of the pool walls, water covering the core, and materials in the core itself.

The source of gamma rays includes both prompt gammas from the fission of U-235, and decay gammas from the fission product inventory in the core. Neutrons will also result from U-235 fission and delayed neutron emitters in the fission product inventory. The intensity of the prompt neutron and prompt gamma is proportional to reactor power. The intensity of the delayed neutrons and decay gammas is a function of the operation history of the core and, if the reactor is shutdown, the elapsed time between the time of observation of the shutdown field and the time of the last reactor shutdown.

11.2.4.1 Direct Exposures

Personnel exposures to radiation emanating directly from the core can occur in several ways. Gammas and neutrons from the core can penetrate the water covering the core or the walls of the pool and result in a dose. A person can also be exposed to core radiation through an opened experiment facility. These cases will be considered in this section.

11.2.4.2 Gamma Dose from the Core through Pool Water

The gamma dose from the operating core at 250 kW was conservatory estimated to be 14.5 mR/hour. Using build-up and 4 energy groups, and the following equations.

$$\phi_b = \frac{S}{4\pi R^2} B_p(\mu R) e^{-\mu R}$$

$$\dot{X} = 0.0659 \sum_i I_i E_i \left(\frac{\mu_a}{\rho} \right)$$

11.2.4.3 Gamma Dose Through Unrestricted Areas

The gamma dose to the room directly above the reactor would be approximately 0.1 mr/hour. Therefore the reactor can operate 1000 hours per year. Well above the current usage. Dose estimations were done assuming 10 feet of air to the ceiling and 1 foot of concrete floor.

11.2.4.4 Exposure From Experimental Facilities

Several experimental facilities are located adjacent to or within the core. These include the dry tube, the pneumatic transfer or rabbit system, the central irradiator, and the fast flux irradiators. The following section names and describes the various experimental facilities and presents conservative dose estimates for potential radiation derived from each facility.

Dry Tube Irradiator:

The dry tube is a hollow tube and sits to the side of the core. The dry tube follows a straight path part way up the pool, and then slowly curves until it is approximately at a 45 degree angle at the top of the pool. Because of this curvature in the dry tube, there is no direct radiation from the core. however in order to estimate the dose rate at the exit of this dry tube, scattering was neglected. First, a measurement of the dose was made during 90 kW operation. At an operating power of 90 kW, the measured dose is at or below background radiation and no extrapolations can be made from the 90 kW data. However using the assumptions outlined in Appendix B conservative (assuming no scattering is estimated of the dose rate is 0.485 mR/hr. See Appendix B, Section 3 for details of these calculations.

The Pneumatic Transfer Tube:

The pneumatic transfer tube is inserted directly into the core of the reactor, extending upwards to within 2 m below the surface of the pool and then the tube curves gently. Due to the geometrical similarities with the dry tube, the dose rate estimation for the pneumatic transfer tube is a 0.485 mR/hr (assuming no scattering).

The Central Irradiator:

The central irradiator is an empty fuel rod that is placed in the center hole of the core grid. There is no direct radiation from the core.

The Fast Flux Irradiator:

The fast flux irradiators are aluminum casks that are lowered down to the core and placed into slots that sit next to the core. Again, there is no direct radiation path from the core to the top of the pool through this facility.

Since there is no direct path from any of the experimental facilities to the top of the pool, the gamma dose received from the maximum dose from the experimental facilities is 0.97 mr/hr.

11.3 Radiation Protection Program

Production and use of radioactive materials within the reactor lab are subject to the guidelines issued by the University's Radiological Health Office. These guidelines in turn fall within the regulatory framework of 10 CFR 20 and provide for more stringent controls than those specified in these regulations in most cases. In addition, the CENTER follows internal procedures that fall within the guidelines of the University of Utah, the Utah State Division of Radiation Control, and federal regulations.

11.3.1 ALARA Program

The approach to radiation protection at the CENTER is to keep radiation exposures to personnel within the constraints of the ALARA (As Low As Reasonably Achievable) program. This includes the use of methods and procedures involving shielding of radiation sources and/or personnel, increasing the distance between an exposure point and a radiation source, reducing the time a person might be exposed to a given dose rate, containment of sources and careful, thoughtful, advanced planning when working in an area which might contain a radiation field.

Various administrative controls have been implemented at the CENTER to compliment the ALARA approach. All experiments involving the reactor are reviewed by a Senior Reactor Operator, licensed for the UUTR TRIGA reactor. Requests for reactor use that involve radioisotope production or any other potential radiation exposure of significance are examined for indications or estimates of radiation hazard, and methods and/or procedures are developed as needed to reduce the potential hazards. If a radiation accident is possible, an experimenter must provide a methodology for dealing with such an event and steps to be taken to mitigate its consequences. All experiments that use the TRIGA nuclear reactor are subject to the approval and concurrence of the Senior Operator on duty during the reactor operation. If this individual determines that a particular reactor operation poses a hazardous or potentially unsafe condition, the run may be terminated at his or her discretion.

The radiation monitoring and surveying program is structured so that all categories of radiation sources (air, liquid, and solid) are detected and assessed in a timely manner. To accomplish this the monitoring program is broken into two categories, these being routine and surveillance surveys. The distinction between the two being that routine surveys are conducted by the laboratory staff in conjunction with operations and normal inspections, and surveillance surveys being conducted by an independent agency (Radiological Health) in support of general compliance

11.3.2 Radiation Monitoring - Routine

Several systems are employed to support the routine radiation monitoring of the UUTR facility. These are the area radiation monitoring system, the continuous air monitor, the auxiliary radiation monitor, monthly surveillance surveys, and routine handling surveys.

The area radiation monitoring system (ARM), consisting of four detectors placed in strategic locations, provides radiation surveillance for the CENTER TRIGA reactor facility. The detectors are located in the most likely areas of uncontrolled release of radiation. One is positioned at the top of the reactor pool to detect the radiation at pool level. Another detector is positioned on the ceiling above the reactor to monitor the conditions of the room and for persons on those levels above the reactor (floors 2 and 3.) A third detector is in the ventilation duct and is used to determine the activity leaving the building through the ventilation system. The fourth detector is located in the counting lab to monitor personnel at the PTS terminus.

The continuous air monitor (CAM) is attached to the facilities ventilation system. This monitor is designed to detect iodine, noble gas, and particulate radiation levels in the air. The CAM is the first instrument that will detect a leaking fuel element during reactor operation. This instrument only functions during reactor operation.

All read-outs, indicators, and alarms associated with the ARM and the CAM can be monitored from the main control console. The emergency ventilation system is operated by the detector at the top of the reactor pool. The system actuates an audible and visible alert at alarm levels. When the levels are exceeded, the emergency ventilation system will be activated.

During all operations an auxiliary radiation monitor is required for operation. This detector serves as back-up for operations such that even if the ARM system fails the radiation levels are still monitored. This detector is only used during reactor operation.

On a monthly basis the UUTR staff conducts radiation level surveys in the reactor room storage area. The philosophy behind this procedure is that to control exposure, the potentially strong sources must be characterized, as well other areas that might go unnoticed otherwise. In addition on a monthly basis a water sample is taken from the reactor pool and tested in a HPGe for the presence of fission products.

Surveys are also performed after routine handling of radioisotopes, work in the vicinity of radiation sources, and of recently irradiated materials. These surveys are performed to ensure that no contamination has passed from an object to a handler and that any irradiated objects radiation fields are well characterized.

11.3.3 Radiation Monitoring - Surveillance

On a monthly basis the University of Utah's Radiological Health Department conducts radiation surveillance throughout the facility. Radiological Health performs radiation level surveys, contamination wipes, and monitors dosimetry placed throughout the facility. In several locations in each room several radiation level measurements are made. These results are posted throughout the UUTR facility at each doorway. Several radiation level measurements are performed outside the facility near the facilities strongest sources in order to ensure no exposure in uncontrolled areas. In addition to conducting radiation level surveys, wipes are made in the same locations as those where the radiation levels are taken. These wipes are then measured for both alpha and beta contamination. Film and TLD badge dosimetry is also used for gamma and neutron doses throughout the facility. A series of badges are mounted at various points in the main reactor room, the control room, and the rest of the reactor lab as well as on the floor above the reactor. These badges provide a long-term average of area dose in the vicinity of the badges.

11.3.4 Survey Instruments

Several portable survey instruments are available at the CENTER for routine monitoring of radiation sources. These survey instruments provide enough range and flexibility to monitor most expected radiation sources produced by reactor operation. For those sources not detectable with these instruments, the Radiological Health Office can provide additional support. The instruments available for routine monitoring and surveys are listed in Table 11.3.4

Table 11.3.4

Instruments Available for Routine Monitoring and Surveys

Instrument	Location	Function
Area Radiation Monitors (4)	-Stack Effluent Monitor -Reactor Room Tank -Reactor Room Ceiling -Counting Room	Measure Radioactivity In Stack Effluent, And Measure Gamma Radiation Fields In Reactor and Counting Room
Continuous Air Monitors (3)	-Stack Effluent Monitors	Measure Radioactivity In Stack Effluent
Micro R Meter (1)	-Control Room	Auxiliary Meter During Operation, And Monthly Radiation Level Surveys
Portable GM Survey Meter (4)	-Reactor Room -Control Room	Personnel Contamination Survey
Pancake GM Survey Meter (1)	-Reactor Room	Beta Gamma Dose Rates
Ion Chamber (1)	-Reactor Room	Beta Gamma Dose Rates
Pressurized Ion Chamber (1)	-Facility Entrance	Emergency Survey Meter
Portable Alpha Detector (1)	-Counting Laboratory	Personnel Contamination Survey
Liquid Scintillation Detector (1)	-Counting Laboratory	Personnel Contamination Survey
Neutron Detector (1)	-Reactor Room	Measure Neutron Dose Rates
HPGe (2)	-Counting Laboratory	Gamma Spectroscopy
Portable NaI	Counting Laboratory	Gamma Spectroscopy
Air Flow Velocity meter (1)	-Control Room	Measure Air Flow Rate

11.3.5 Radiation Exposure Control and Dosimetry

All persons working in the reactor lab are trained in appropriate radiation protection concepts. In addition, these persons are trained to assist with the emergency response to abnormal radiological conditions in the reactor lab. Certification of this training is kept on file at the CENTER, as required by 10 CFR 19.

Occasional visitors to the reactor lab such as commercial vendors, one-time visitors for tours and demonstrations, and non-routine experiment personnel, are given basic instruction in wearing dose monitors, signing in and out of the lab, and radiation and radioactive material storage areas. These persons are escorted by a member of CENTER staff.

11.3.5.1 Access Control

The control room and the reactor room are designated as restricted areas. Outer doors are locked to prevent unauthorized entry. Access to the reactor facilities is tightly controlled Only personnel trained in radiation protection and security procedures are issued room keys. All persons must enter these rooms through the inner doors with a key or accompanied by an authorized individual who has been issued a key. Upon entering the restricted area, an individual without key access must complete the sign-in procedure and either wear a dose monitor or be accompanied by an individual wearing a dose monitor at all times.

Occasionally, areas may be posted as radiation areas; and, in rare cases, an area may be posted as a high radiation area. Appropriate restrictions and precautions are observed in all such cases.

The sign-in procedure required for access to the reactor lab documents specific information from all persons admitted to the facility. Visitor's cards are maintained as part of the permanent records of the reactor lab.

11.3.5.2 Personal dosimetry

The CENTER staff, faculty, and students that work in the reactor lab on a regular basis are required to wear a TLD badge. Badges used at the reactor lab contain beta, gamma and neutron sensitive materials for monitoring doses at various tissue depths. Designated personnel wear ring badges for monitoring extremity doses. The ring badges are also TLD badges that contain gamma sensitive materials. The regular badges and ring badges are both read on a monthly basis. All visitors are given a direct read dosimeter and follow the procedures outlined in section 11.4.5.1. Several dosimeters are available in the reactor lab.

11.3.5.3 Shielding

Earlier sections discussed the radiation shielding features of the TRIGA reactor. In addition, lead bricks are placed in strategic locations within the reactor lab to supply the necessary shielding. Smaller lead containers, called "pigs", are available for storage of small radioactive sources. These lead "pigs" vary in wall thickness from a fraction of an inch up to several inches and are used routinely.

11.3.5.4 Administrative Controls

Safe operation of the TRIGA nuclear reactor and performance of associated experiments depends on reliable and conscientious observance of established procedures and protocols by the staff of the CENTER. Experiments involving production of radioisotope sources of significant quantities and intensities, use of experimental facilities such as the dry tube, and the use of radiation sources apart from the reactor, are subject to the approval of appropriate CENTER or UUTR Reactor Safety Committee. Such experiments and uses must be in accordance with regulations specified in 10 CFR 20 or other applicable regulations. Administrative controls are established to allow CENTER personnel discretion in approving and carrying out experiments and operations in a safe manner. CENTER personnel may themselves specify appropriate procedures to be incorporated in an experimental program to assure radiological safety. If such procedures are not observed during the experiment, the Senior Reactor Operator on duty during the operation may terminate the experiment at his or her discretion.

11.3.5.5 Contamination Control

Radioactive contamination is controlled at the UUTR by using written procedures for radioactive material handling, by using trained personnel, and by operating a monitoring program designed to detect contamination in a timely manner. The following methods limits contamination in the CENTER.

1) procedures and training developed to limit and control contamination during routine sample handling and reactor maintenance.
2) anti-contamination clothing used when appropriate.
3) all new personnel must be trained by Radiological Health prior to working in the facility.
4) contamination events documented and then discussed in staff meetings (this practice helps avoid repeating events which caused contamination)
5) all samples encapsulated prior to irradiation to help minimize the potential for contamination.

11.3.5.6 **Environmental Monitoring**

The environmental monitoring program at the University of Utah monitors both gamma dose rates in and out of the facility, and effluent releases. Effluent releases can be broken down into two categories; gaseous and liquid. As stated in Ar-41 release is closely monitored and exits through a single exhaust stack. Records of all releases through the exhaust stacks are maintained. Liquid releases are disposed of in accordance with regulations by University of Utah and 10 CFR 20 "release to sanitary sewer". All releases are accompanied by written procedure and radioactive disposition record. Environmental dosimeters are placed in six locations off site.

11.4 Radioactive Waste Management

Radioactive waste generated by operation of the UUTR reactor will be evaluated to determine the isotope(s) present and their respective half-lives. After characterization the waste is transferred to the University of Utah's Radiological Health department for disposal

11.4.1 Radioactive Waste Management Program

The objectives of the UUTR radioactive waste program are to minimize waste and to properly store, sort, and handle the waste. The University of Utah's Radiological Health department is responsible for the administration of radioactive waste disposal for the reactor facility. Specific procedures for the disposal of radioactive waste are outlined in Radiation Procedures and Records #54. Waste management and radionuclide transportation training are both part of the initial and specialized training of all personnel at UUTR. This training is administered by Radiological Health. In addition to the training and administration of radioactive waste management, Radiological Health maintains all records of radioactive wastes in a database.

11.4.2 Radioactive Waste Storage

The CENTER does not store waste. Samples, gloves, and sources that are no-longer useful are characterized and transferred to Radiological Health for disposal.

11.4.3 Radioactive Waste Controls

At UUTR radioactive wastes are generally considered to be any item or substance which is no longer of any use to the facility and which contains or is suspected of containing, radioactivity above the established natural background radioactivity. Because UUTR waste volumes are small and the nature of the waste items is limited and reasonably repetitive, there is usually little question about what is or is not radioactive waste. Equipment and components are categorized as waste by the reactor operations staff, while standard consumable supplies like plastic bags, gloves, absorbent materials, disposable lab coats, etc., automatically become radioactive waste if detectable radioactivity above background is found on any of these items.

When possible, radioactive waste is initially segregated at the point of origin from items that will not be considered waste. Screening is based upon the presence of detectable radioactivity using appropriate monitoring and detection techniques and on the projected future need for the items and materials involved. All items and materials initially categorized as radioactive waste are monitored a second time before packaging for disposal to confirm data needed for waste records, and to provide a final opportunity for decontamination/reclamation of an item. This helps reduce the volume of radioactivity by eliminating disposal of items that can still be used.

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Chapter 12

Conduct of Operations

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List of Figures and Tables

(Figures and Tables are numbered according to the sections that refer to them.)

12 CONDUCT OF OPERATIONS

This chapter discusses the conduct of operations at the UUTR reactor facility. The conduct of operations involves the administrative aspects of facility operation including facility organization, review and audit activities, organizational aspects of radiation safety, facility procedures, required actions in case of license or technical specifications violations, reporting requirements, and record keeping. The conduct of operations also involves the facility emergency plan, the security plan, the quality assurance plan, the reactor operator requalification plan, the startup plan, and environmental reports.

12.1 Organization

This section will discuss organizational structure, and the responsibilities, selection and training of personnel at the UUTR facility. The purpose of this section is to demonstrate that the management and staff of UUTR are knowledgeable about the technical requirements for operating a safe facility, are in compliance with regulations and license conditions, and will implement an effective radiation protection program to protect the health and safety of the public, the facility users, and the staff.

12.1.1 Structure

The administrative organization of the UUTR is summarized in Figure 12.1.1. The lines of responsibility and the lines of communication (consultation) are to be strictly followed. Failure to follow this organizational structure is a reportable occurrence as defined in TS 1.1. The organizational chart is essentially a set of three columns and three rows. The columns are defined by common functions: operations, reactor safety, and radiation safety. The rows are defined by levels of responsibility: administration, management, and staff. Administration is responsible for the overall operation of the facility including coordination of operations, reactor and radiation safety sections, budget authorization, and interfacing with regulatory organizations. Management is responsible for assisting administration when necessary and supervising day to day operations. Staff are responsible for executing day to day operations tasks.

The UUTR maintains a small staff of Reactor Operators(RO) and Senior Reactor Operators(SRO). There are no specialized staff solely responsible for implementing the radiation safety function, but each staff member has been trained in radiation safety and is responsible for the implementation of a radiation protection program. The direct line of responsibility for operation of the reactor is as follows: RO's, SRO's, Reactor Supervisor (RS), and Reactor Administrator, President (U of U), and the Board of Trustees.

The Department of Radiological Health and the Reactor Safety Committee are separate support entities, which provide independent radiation surveys and audits, respectively, for the UUTR. Both of these committees are directly responsible to the University of Utah Vice President for Research and not to management or staff of the UUTR; thus the independent characteristic of radiation safety reviews and audits is guaranteed.

12.1.2 Responsibility

The top officials of the administration of the University of Utah including the Vice President for Research, the President, and the Board of Trustees are responsible for overseeing the operations and general maintenance of the UUTR.

The Reactor Safety Committee (RSC) is responsible for reviewing proposed experiments, modifications and procedures, and changes thereto, with respect to the TS, CFR, SAR, and ANSI Standards, the Experiment Authorization and Modification Authorization procedures established and general common sense. The RSC is also responsible for auditing operation, operational records, any operating abnormalities, and the expected performance of facility equipment effecting nuclear safety. The RSC makes recommendations to UUTR through the appropriate channels based on their reviews of these items. In addition, the RSC reviews any reported safety-limit violations and assists in preparation of required reports, as necessary.

In order to fulfill these responsibilities, the RSC shall meet at least semi-annually and on call of the Chairman. Documentation of the activities of the RSC shall be maintained. The documentation shall include the names and qualifications of the members, the agenda and approved minutes of all RSC meetings, RSC actions, and copies of all correspondence and reports to or from the RSC. RSC documentation is transmitted annually to the University Archives.

The Radiation Safety Committee, appointed by the President of the University, is responsible for radiological safety on the entire university campus and controls the movement and use of all radioisotopes and radiation producing machines on campus. The RS in full consultation with the RSO enforces the regulations of this committee within the reactor area.

The RA is responsible for liaison with the NRC regarding technical and emergency matters and for enforcement of all regulations. The RA has final authority and ultimate responsibility for the reactor facility and, within the limitations established by the facility license, makes final policy decisions on all phases of reactor operation, appoints personnel to all positions that report to the RA, and is advised in matters concerning radiation safety by the Radiation Safety Committee and matters concerning safety by the Reactor Safety Committee. This individual holds a Senior Reactor Operator's License.

The RS's responsibilities include enforcement of administrative rules and operating procedures, all planning and scheduling, the reviewing and classifying of proposed experiments, direction of reactor operators and trainees, and maintenance of reactor operation and RSC activity records. The RS also consults with the RSO and the Radiation Safety Committee concerning radiation safety aspects of facility operation, enforces compliance within the reactor facility with radiological requirements, monitors operations and is responsible for the operation, maintenance, and scheduling of the neutron generator facility and all other radiation sources in the laboratory. The RS may approve changes in procedures (see TS 6.8) when necessary. The RS may perform any duties not specified and those duties specified for RO's and SRO's.

The RSO is an experienced health physicist and is responsible for the radiological health and safety of the University community. The RSO works with the RS to ensure the radiological safety of operations within the reactor facility and is responsible for the transfer of radioisotopes outside the CENTER. The RSO is available for consultation in the event of any emergency. The RSO may perform only those duties specified for the RSO or a health physicist.

Activities in the facility will always be under direct control of an NRC licensed SRO designated by the RS. An SRO must be on call (but not necessarily on site) when the reactor is not secure. The SRO will be responsible to the RS for the overall facility operation including safe operation and maintenance of the facility and its associated equipment. Temporary changes to procedures that do not change their original intent may be made by the SRO (see TS 6.8). The SRO may perform any duties not specified here and duties specified for RO's. RO's licensed by the NRC can legally operate the TRIGA reactor. RO's may directly supervise trainees when manipulating core reactivity. All RO's and trainees are under the direction of an SRO.

12.1.3 Staffing

The UUTR currently has two NRC licensed Senior Reactor Operators. There are no Reactor Operators at present. RO's and SRO's must have a graduate degree in Nuclear Engineering (or a related field) or be working towards the completion of a graduate degree. At the current staff level, the UUTR is in full compliance with the requirements of 10 CFR 50.54.

12.1.4 Selection and Training of Personnel

The Reactor Administrator and Reactor Supervisor must be licensed Senior Reactor Operators. In addition, they must hold a current academic or research faculty appointment at the University of Utah. RO's and SRO's must have a graduate degree in Nuclear Engineering (or a related field) or be working towards the completion of a graduate degree.

The UUTR maintains a RO/SRO qualification and requalification program to ensure the competence of its operators. The training program covers basic nuclear phenomena, health physics, and reactor operations and involves both lectures and hands-on experience. The requalification program presents the same material. All licensees must take part in an annual requalification exam. The Reactor Supervisor is responsible for conducting the training and requalification programs.

12.1.5 Radiation Safety

The radiation safety program at the UUTR consists, in part, of monthly monitoring activities performed by the CENTER staff and the University of Utah Department of Radiological Health. Radiation analysts with extensive training in the area of health physics perform radiation surveys of designated laboratory surfaces as well as outer walls of the laboratory, where there is unrestricted access to a large amount of pedestrian traffic. These analysts report directly to the Director of Radiological Health who is the Radiation Safety Officer(RSO) for the University of Utah. The RSO sits on both the Radiation Safety Committee and Reactor Safety Committee. The RSO, with the approval of the aforementioned independent committees, can interdict or terminate safety-related activities if necessary. Each month a copy of the survey results is provided to the Reactor Supervisor and filed.

In addition, the radiation safety program includes the wearing of personal dosimetry, i.e. TLD badges and ring badges, by all UUTR staff who work in the laboratory. The Department of Radiological Health sends the badges to an independent laboratory to be analyzed. The results are provided to the Reactor Supervisor and filed. The Radiation Safety Officer reports dosimetry results and survey results to the Reactor Safety Committee each quarter.

12.2 Review and Audit Activities

An independent oversight committee referred to as the Reactor Safety Committee (RSC) conducts the review and audits on a biennial basis. Appointments to the Reactor Safety Committee are initiated by the RSC itself and are annually reviewed and approved by the President of the University.

The Reactor Safety Committee has been assigned approval authority for review and audit activities. The RSC communicates with UUTR management through the members of the UUTR team who are part of the committee. The RSC reports to the Office of the Vice President for Research at the University of Utah.

12.2.1 Composition and Qualifications

The committee shall be composed of no less than five members and must consist of a Chairman, the Reactor Administrator, the Reactor Supervisor, the Radiation Safety Officer, and one additional member not involved in reactor operations but knowledgeable in fields related to nuclear reactor safety. It is preferable to have one additional member who represents UUTR operating staff, and one additional member from outside the university who is knowledgeable in fields related to nuclear engineering and health physics. The qualifications of current committee members are maintained at the UUTR.

12.2.2 Charter and Rules

To conduct the business of the committee, a quorum must be present at the meeting. A quorum consists of a majority of its members and must include the Chairman, the Reactor Supervisor, and the Radiation Safety Officer. In addition, an assembly of RSC members cannot constitute a quorum if the majority present are UUTR staff.

The RSC shall meet at least semiannually, and a subcommittee thereof shall meet at least quarterly. The committee shall also meet upon the call of the Chairman. Prior to the meeting, an official agenda with copies of all proposals (e.g., new experiments, proposed amendments to licenses or technical specifications, etc.) for committee review shall be distributed to all members and shall constitute the principal business at each meeting.

Robert's Rules of Order shall establish conduct in all meetings of the Committee. The regular order will be: (1) Roll call; (2) Report by the RA; (3) Report by the RS; (4) Report by the RSO; (5) Reports by other individuals or subcommittees; (6) Approval and corrections of the minutes of the preceding meeting; (7) Approval of experiments, procedures etc.; (8) Review of communications, audit of records and reactor operations, logs and miscellaneous matters; (9) Selection of date for the next meeting.

12.2.3 Review Function

The following section introduces the review function of the Reactor Safety Committee. The nature and frequency of mandatory reviews performed by the Reactor Safety Committee are further described in Chapter 14 of this document.

The committee reviews and approves all aspects of the reactor facility associated with safety in conformance with NRC and University regulations. This includes review and approval of all proposed new experiments and procedures as required, and determination if proposed experiments, procedures or modifications involve unreviewed safety questions, as defined in 10 CFR 50, Part 50.59 (c), or conflict with the written Technical Specifications. The committee must also review all reported abnormal occurrences and violations of the Technical Specifications, evaluate the causes of such events and the corrective actions taken, and recommend measures to prevent recurrence.

12.2.4 Audit Function

The Reactor Safety Committee audits reactor operations semiannually at intervals not to exceed eight months. The semiannual audit includes review of the reactor operating records (reactor operations log, maintenance log, surveillance and procedures log, startup and terminations log, fuel procedures and log, core procedures and log, and experiments log), inspection of the reactor operating areas, and radiation exposures at the facility and adjacent environs using the data collected by dosimetry for personnel and environmental monitoring. If necessary, the RSC will also audit any unusual or abnormal events.

The audits are currently performed by volunteer members of the Reactor Safety Committee who are not operating staff of the UUTR, the Reactor Administrator, Reactor Supervisor.

12.3 Procedures

The utmost care has been taken to keep thorough, accurate records of UUTR operations, including detailed documentation of standard operating procedures for all routine activities and the methods of review for proposed and existing procedures.

These documents, known collectively as the "CENTER Surveillance and Procedures Log" apply only to activities performed in the CENTER at the U of U. Facility users are directly accountable for any activities they may undertake within the facility which are not outlined in these documents. This section lists the forms and procedures included in the "CENTER Surveillance and Procedures Log" as well as the approval process for new standard procedures. It also describes UUTR policy in the event that a deviation from standard procedure is necessary during reactor operation.

Each form is numbered and named according to the procedure that it documents. Table 12.3 is a current list of all approved CENTER procedures Pursuant to 10 CFR 50.59, changes to the procedures described in this safety analysis report may be implemented providing that all changes are documented, no conflict with the facility Technical Specifications exists, and the change does not involve an unreviewed safety question. UUTR operating staff, including the Reactor Supervisor, must create a form for documenting each new standard procedures or make revisions to existing forms as necessary. The Reactor Safety Committee must approve all new forms and all revisions. Each approved form indicates the date of approval on the top right side of the form, and the signature of the RSC Chairman is located across the bottom of the last page on every form. Temporary changes during reactor operation may be made by a licensed SRO to deal with special and unusual circumstances. These changes shall be documented and subsequently reviewed by the Reactor Safety Committee.

12.4 Required Actions

This section of the UUTR Safety Analysis Report discusses the actions taken in the occurrence of a reportable event or a violation of the facility safety limits. In addition, the circumstances that constitute a reportable event are defined.

The facility safety limits are established in the facility Technical Specifications and Chapter 14 of the SAR document. A reportable event is the occurrence of any of the following events during reactor operation: operation with any safety-system setting less conservative than specified by the Technical Specifications; operation in violation of a limiting condition for operation; operation with a required reactor or experiment safety-system component in an inoperative or failed condition which could render the system incapable of performing its intended safety function; any unanticipated or uncontrolled change in reactivity greater than $1.00; poor application of administrative or procedural controls, possibly compromising specified safety-limits; and, lastly, a measurable release of fission products into the environment.

In the event that a safety limit is exceeded or in response to a reportable event, the following actions are required: the reactor shall be shut down, and reactor operation shall not be resumed until authorized by the U.S. Nuclear Regulatory Commission (NRC); an immediate report of the occurrence shall be made to the Chairman of the Reactor Safety Committee, and reports shall be made to the NRC in accordance with facility Technical Specifications; a report shall be prepared that shall include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence; this report shall be submitted to the Reactor Safety Committee for review and then submitted to the NRC when authorization is sought to resume operation of the reactor; in addition, all reporting requirements established by the facility Technical Specifications, and described in Section 12.5 of this document are applicable.

12.5 Reports

This section discusses the content and frequency of reports to the Nuclear Regulatory Commission on either a regular basis or as required by special or unusual circumstances.

An annual operating report must be submitted to the Nuclear Regulatory Commission (NRC) within 60 days following the 30th of June each year. The document should be submitted both to the NRC document control center. This report shall include the following information: a brief narrative summary of operating activity, changes in facility design, performance characteristics, and operating procedures related to reactor safety and occurring during the reporting period, and results of surveillance tests and inspections; tabulation of energy output since initial criticality; the number of emergency shutdowns and inadvertent scrams, including reasons for them; discussion of the major maintenance operations performed during the period, including the effect, if any, on the safety of the operation of the reactor and the reasons for any corrective maintenance required; a brief description of the safety evaluations of changes in the facility or in procedures, tests and experiments carried out pursuant to 10 CFR 50.59; a summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or before the point of such release or discharge, summarized on a monthly basis and including liquid, gas, and solid waste forms; an annual summary of the radiation exposure received by facility personnel and visitors; an annual summary of the radiation levels of contamination observed during routine surveys performed at the facility in terms of the average and highest levels; and an annual summary of any environmental surveys performed outside the facility. The UUTR Technical Specifications describe the requirements of the annual operating report in greater detail.

Special reports shall be made to the NRC in case of the following conditions: any reportable event; any violation of a safety limit; or any accidental release of radioactivity above permissible limits in unrestricted areas whether or not the release resulted in property damage, personal injury, or exposure. In each case listed, an initial report by USNRC Operations Office. This must be followed by a written report submitted within 10 days to the NRC Document Control Center and USNRC Operations Office in Washington, D.C.

A written report must be submitted to the NRC Document Control Center and a copy to the USNRC Operations Office within 30 days of the following occurrences: any significant variation of measured values from a corresponding predicted or previously measured value of safety-connected operating characteristics occurring during operation of the reactor; any significant change in the transient or accident analysis as described in the Safety Analysis Report; any significant changes in facility organization; or any observed inadequacies in the implementation of administrative or procedural controls.

Upon receipt of a new facility license or an amendment to the existing license authorizing an increase in reactor power level, a report must be submitted within 60 days to the NRC Director of the Office of Nuclear Regulation. The purpose of this report is to document the measured values of the operating conditions. The report must include an evaluation of facility performance to date in comparison with design predictions and specifications, and a reassessment of the safety analysis submitted with the license application in light of measured operating characteristics when such measurements indicate that there may be substantial variance from prior analysis.

12.6 Records

This section of the SAR discusses the record keeping system for the UUTR and includes information about the types of records maintained at the CENTER and the period of retention for these records.

All CENTER forms (listed in Table 12.3) completed within the current calendar year are stored in loose-leaf notebooks. Forms completed before the current calendar year are stored in folders within a filing cabinet for a minimum retention of three years. This cabinet is located within the UUTR control room, and it is locked when records are not being used. The filing cabinet is well organized, and contains top-to-back and alphabetical directories on the first file of the topmost drawer. After the three-year retention period, the completed forms are removed from the cabinet and archived with University of Utah Records Management. All archived records are carefully catalogued (storage records on file at the CENTER and at Records Management Department) and can be accessed within 24 hours.

In addition to operations records, the control room filing cabinet also holds facility documents, NRC correspondence, and training records which includes the facility license, all current operator's licenses along with requalification documents and medical certifications, emergency plan, physical security plan, current Safety Analysis Report and Technical Specifications, and the University of Utah Office of Radiological Health monthly survey reports for both personnel and environmental monitoring.

12.7 Emergency Planning

This section contains a brief overview of the CENTER Emergency Plan. The Emergency Plan is reviewed every other year and updated during the review if necessary. The objective of the plan is to provide a basis for action and to identify personnel, material resources and designated areas of responsibility for dealing with any CENTER emergency.

The plan contains a description of the reactor, CENTER laboratory, and confinement structure, and identifies escape routes and the location of emergency equipment such as fire extinguishers, first aid kits, and an evacuation horn. The plan also identifies both on-site and off-site support organizations that shall be contacted for assistance if required, such as the Salt Lake City Fire Department, Campus Security and University Police, Gold Cross Ambulance, and the University Hospital. The organization and responsibilities of CENTER personnel and of support organizations are described in detail. The plan stresses the requirement that CENTER operating staff maintain control of all emergency response actions. All emergency response organizations shall be directed by the Reactor Supervisor or, if this individual is absent, the designated Senior Reactor Operator present at the time of the emergency.

The plan calls for an annual emergency training at the CENTER. All emergency response organizations that may be requested to assist in the event of an emergency send representatives to tour the CENTER laboratory, receive information regarding the types of emergencies that can occur and the appropriate responses. During the emergency training, preparations are made to perform the CENTER's annual emergency drill. The instructions for performance, review and documentation of the CENTER emergency drill are described in the plan.

The plan also contains the procedure for classification of a radiological emergency. The only emergency scenarios credible for the CENTER facility can be classified as either Non-Reactor Safety Related Events or Unusual Events. The appropriate response to each emergency class are included in the plan. A set of specific emergency response procedures for a variety of scenarios, which fall under the above two classifications, are attached in an appendix to the plan.

12.8 Security Planning

The CENTER has devised a physical security plan to ensure the protection of special nuclear materials on the premises. All such material associated with the UUTR NRC License R-126 is classified in the category of low strategic significance.

12.9 Quality Assurance

In accordance with Regulatory Guide 2.5 and ANSI 402, "Quality Assurance Program Requirements for Research Reactors," Section 2.17, the UUTR is not required to prepare quality assurance documentation for the facility in the "as-built" condition. This section describes the managerial and administrative controls that ensure the continued safe operation of the existing UUTR.

The principal quality assurance mechanism for the safe operation of the CENTER is the audit function described in Section 12.2.4 of this document and in the Technical Specifications and Audit and Review Plan for the CENTER. This function shall be performed and documented as specified in the aforementioned guidance documents in fulfillment of the regulatory requirements of 10 CFR 50.34.

12.10 Operator Training and Requalification

This section of the SAR describes the CENTER Operator Requalification Program. The purpose of the requalification training program is to ensure that all operations personnel maintain proficiency at a level equal to or greater than that required for initial licensing. The required training schedule for operators, and the content of examinations along with grading criteria, are discussed in this section. In addition, the requirements for maintaining active status and the method of reinstating an inactive operator are discussed.

Licensees enter the requalification program on the date the Nuclear Regulatory Commission issues either a new license or a renewal of an existing license. Licensees continue in the requalification program until either the expiration date of the current license or the date upon which the current license is terminated. The program is offered on an annual schedule and consists of lectures, on the job training, and annual written, oral, and console examinations. The console examination evaluates operator performance at the control console of the UUTR.

The lecture program covers the following eight topics to be presented in eight different lectures: Nuclear Reactor Theory, Radiation Control and Safety, Governing Regulations, Reactor Design, Reactor Control and Safety Systems, Reactor Operating Characteristics, Normal, Abnormal and Emergency, Technical Specifications and License. Any operator may be assigned to present a lecture

Each operator must perform licensed functions for at least four hours per quarter to satisfy 10CFR55.53(e). In addition, each operator participates in the annual console examination to demonstrate familiarity with the following activities: Pre-start checks, Start-up, Operational Power and Termination. Each Senior Reactor Operator performs or directly supervises the completion of these activities with the same frequency required of an operator.

The written examination is administered as an open book exam in a controlled area. The operators may only use the CENTER operator's manual as well as paper, pencils, erasers and calculators to complete the exam. The content of the examinations satisfy the requirements of 10CFR55.41 and may include the requirements of 10CFR55.43. The RS and the RA will be responsible for preparing, administering and grading the written exam on an alternating yearly basis for all other licensed operators. The RA and the Reactor Supervisor prepare, administer and grade the written examinations for each other on a biennial basis.

The criteria for grading and the assignment of pass/fail are as follows. For the written evaluation, the licensee is assigned a rating of either SATISFACTORY or UNSATISFACTORY. In order to obtain a rating of SATISFACTORY, the licensee must attain a minimum score of 70% in each section of the examination. If the licensee fails to achieve a SATISFACTORY rating, that licensee is removed from his or her licensed duties and enrolled in an accelerated training program in the deficient area or areas. For the console evaluation, the licensee is also assigned a rating of either SATISFACTORY or UNSATISFACTORY. In order to obtain a rating of SATISFACTORY, the licensee should demonstrate an understanding of the operation of all apparatus and mechanisms. This evaluation is based upon the ease and smoothness with which the operator performs the Pre-start, Start-up, power operation and termination. If the licensee fails to attain a rating of SATISFACTORY, the licensee is removed from his or her licensed duties and enrolled in an accelerated training program in the deficient area or areas.

An operator may be removed from active status by failing to actively perform the functions of an operator for a period of more than three months or by failing to obtain a satisfactory grade on an evaluation exam. 10CFR55.53 (f) outlines the requirements for operator reinstatement. If an operator has not actively performed the functions of an operator for a period of more than three months, he or she shall satisfactorily demonstrate competence before resuming his or her duties. This is accomplished by performing at least six hours of licensed functions under the direction of a licensed operator or Senior Reactor Operator. Upon completion of this activity, the operator is certified for operation by the Reactor Supervisor.

A record is maintained at the CENTER for each licensee and contains a current copy of the licensee's Reactor Operator or Senior Reactor Operator's license, copies of all written examinations administered to the licensee during the requalification period or the licensee's requalification program progress checklist form CENTER-025. This form contains a record of attended lectures, on the job training, written and console examination evaluations, a record of operator reinstatement, and medical examination completion date. Additional forms may be kept in the licensee's record to provide supporting documentation and may include medical examination forms (10CFR55.53 (i)) and license applications or renewals.

12.11 Startup Plan

The UUTR will increse licensed power level to 250 kW(th) if approved by the NRC. The present core configurations will require slight adjustments in the current fuel loading. All modified core configurations will be analyzed and documented using the approved "CENTER Approach to Critical, operating Power levels will increased in steps not to exceed 50 kW (100 kW, 150 kW 200 kW) Determination of Rod worth and Thermal power calibration procedure". In addition the linear, percent and log power channels will be evaluated for a linear response from the chambers if required the chambers will be moved further from the core. The range switch will also be adjusted to the range setting of 300 kW. The scrams for temperature, and power will be rest the conditions outlined in Chapter 14. The USNRC Operations Office will be notified when the new core is brought on line.

12.12 Environmental Report

In accordance with NUREG - 1537 Part 1, for license renewal and license amendment an Environmental Assessment has been performed. The Environmental Assessment performed on the UUTR for license is included with this re-licensing package.

Table 12.3
Approved Center Forms

Form No.	Title	RSC Approval
Monthly		
CENTER-001R10	TRIGA Prestart Checklist (3 sheets)	04/02/04
CENTER-020R12	Monthly Inspection Checksheet (3 sheets)	04/02/04
Semi-Annual		
CENTER-003R6	Semi-Annual Control Rod Calibrations (2 sheets)	03/29/00
CENTER-011R2	Calibration of Temperature Monitoring Channels (4 sheets)	03/12/97
CENTER-012R3	Semi-Annual Thermal Power Calibration (2 sheets)	03/18/98
CENTER-015R3	Emergency Kit Check	09/17/03
CENTER-038R2	Semi-Annual Ventilation System Maintenance and Verification (3 sheets)	03/26/02
Annual		
CENTER-023R4	Annual Maintenance and Calibration of the Area Radiation Monitors (ARMs) and Continuous Air Monitor (CAM) (3 sheets)	12/17/97
Biennial		
CENTER-002R3	Biennial Control Rod Inspection/Control Rod Movement or Repair (2 sheets)	05/23/02
CENTER-004R1	Biennial Fuel Rod Inspection (2 sheets)	12/17/97
CENTER-009R2	Tank Inspection Procedure (2 sheets)	12/17/97
CENTER-010R1	Heavy Water Reflector Element Inspection Procedure (2 sheets)	12/17/97
Unscheduled		
CENTER-005R4	Core Change and Critical Fuel Loading (2 sheets)	03/29/00
CENTER-006R3	Procedure for Changing Filters in the TRIGA Pool Water Refrigeration/Purification System (4 sheets)	12/17/97
CENTER-007	Procedure to Change Central Irradiator	05/25/88
CENTER-008R4	Procedure for Adding Water to the Reactor Tank (2 sheets)	12/17/97
CENTER-013R4	Adjustment of Power Monitoring Channels (3 sheets)	09/18/02
CENTER-016	Agreement for Off-Hours Access	07/27/88
CENTER-017R1	Familiarization Checksheet	05/13/98

Form No.	Title	RSC Approval
CENTER-018	Fuel Element Inventory Sheet	05/25/88
CENTER-021R21	CENTER Emergency Call List ~ revised 06/10/04	NA
CENTER-022R2	Maintenance Log (2 sheets)	09/21/94
CENTER-024	Replacement of Ion-exchange Resin (2 sheets)	11/30/88
CENTER-025	Requalification Program Progress Checklist (2 sheets)	NA
CENTER-027R4	TRIGA Reactor Irradiation Request and Performance (2 sheets)	03/26/96
CENTER-028R1	Experimental Facility Reactivity Worth Determination Procedure	03/12/97
CENTER-030	Cf-252 Irradiation Request and Performance	03/21/90
CENTER-031	Safeguard Event Log	03/21/90
CENTER-032	Liquid Effluent Discharge Authorization (2 sheets)	03/19/92
CENTER-033	CENTER Security Alarms	05/29/91
CENTER-035R1	Audit and Review Program Checklist (2 Sheets)	06/09/93
CENTER-036	Calibration of pH Meter	06/08/95
CENTER-037	Radiological Emergency Classification Checklist	12/14/94

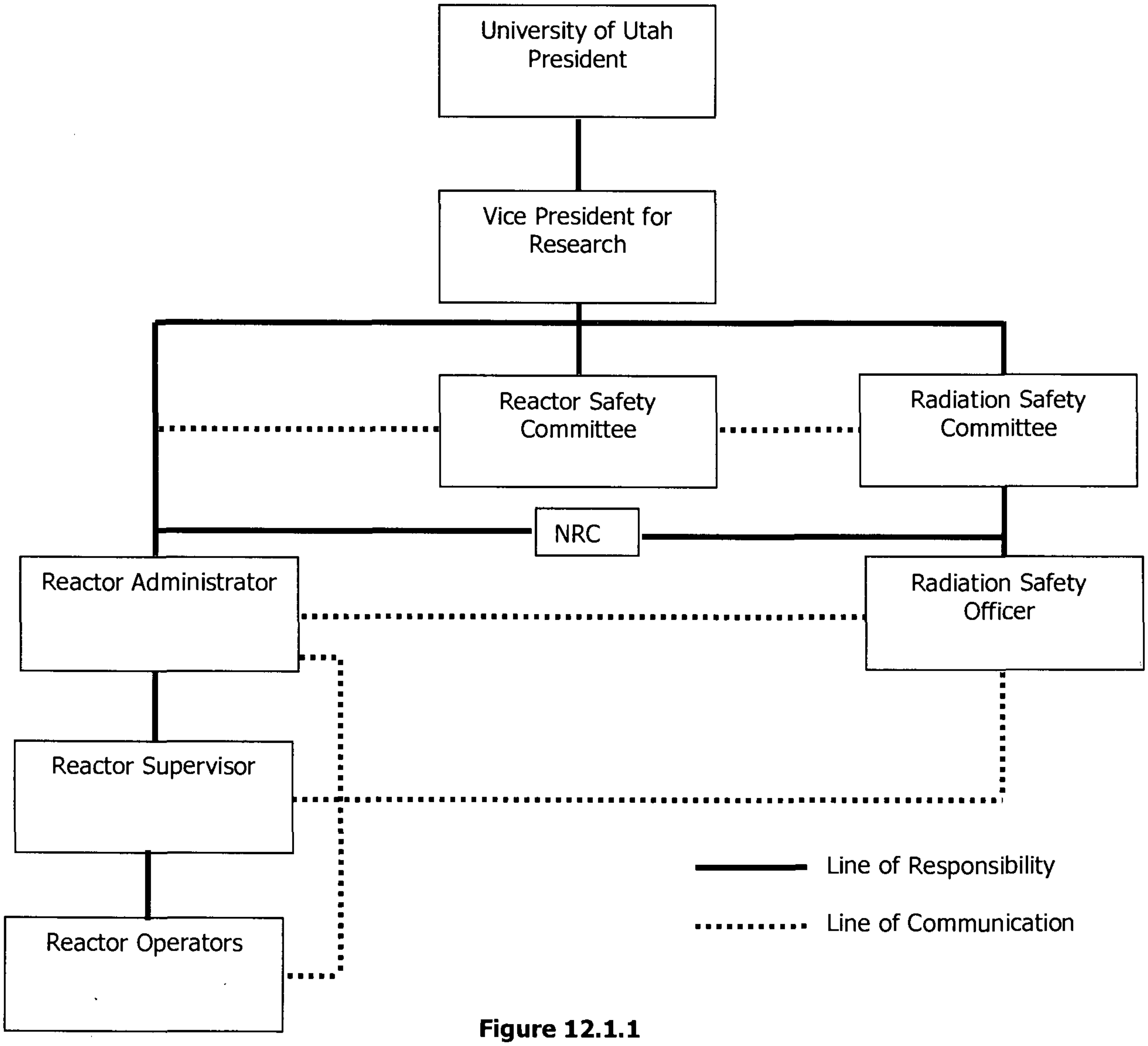

Figure 12.1.1

Administrative Organization Chart

Chapter 13

Safety Analysis

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13 Introduction and Overview

The following section considers in detail the design features of the reactor and building to illustrate that the reactor may be operated under the specified conditions with no hazard to the health and safety of the operator, other occupants of the Merrill Engineering Building or to the general public. The effects of conceivable accidents due to component malfunction, human error or "act of God" events are also considered. The details of these calculations are presented in Appendix C. The following types of accident situations were considered for this safety analysis:

(1) Loss of coolant
(2) Reactivity insertions
(3) Fuel cladding failure
(4) Natural phenomena .

For each situation, a worst case scenario was formulated to ensure more conservative calculations.

13.1 Loss of Coolant

Safety analysis for a loss of coolant scenario was performed for two separate conditions. One condition was a cataclysmic (earthquake) event, where the bottom of the reactor tank was completely opened to the soil and the water was allowed to drain out of the tank. The other condition was the evaporation of the entire contents of the pool through natural convection. Both of these worst-case scenarios are highly unlikely.

13.1.1. Cataclysmic Accident

From the soil properties, which were determined from the geological data, the time to drain the tank through the complete opening of the bottom of the pool was calculated using Darcy's Law. The results from this calculation estimate the time to drain the tank is [redacted] hours. In this amount of time, an emergency water line can be set up to keep the core under water and provisions can be made to prevent or minimize the dangers of uncovering the fuel elements in the core.

13.1.2 Evaporative Coolant Loss

In order to analyze the event of the water makeup line to the tank not functioning properly, calculations were performed to predict consequences of the evaporation of the entire contents of the tank at full power. The minimum time it takes to evaporate 1 kg of water AT 250 kW(th) operation is 9.04 sec. The mass of the pool water is 31,314 kg for the entire tank. The time to evaporate all of the pool water is 18 hours In this amount of time, an emergency water line can be set up to keep the core under water and provisions can be made to prevent or minimize the dangers of uncovering the fuel elements in the core.

13.1.2.1 Radiation From an Exposed Core

The exposure dose rate resulting from a loss of pool water scenario is estimated. The dose rate from the decay power after operating at 250 kW is negligible. The dose rate from the core structures was estimated as 8 R/hour at 18 inches with no shielding. As stated earlier there is sufficient time to accommodate emergency shielding of the core due to loss of water.

13.2 Reactivity Insertion

The sudden insertion of reactivity can cause a rapid change in reactor power. For this analysis, a sudden positive reactivity insertion was assumed. Since negative reactivity insertions cause a decrease in reactor power, which is considered a safe reactivity insertion, this scenario was not considered. The limiting factor in the amount of reactivity that can be safely inserted into the reactor is the fuel temperature. It was determined through Gulf Atomic experiments and analytical heat transfer analyses that the maximum fuel temperature is reached before the maximum cladding temperature is reached.

For the analysis, it was assumed that the reactor was operating at 0 W (shutdown) and 250 kW (full power) when the insertion was made. Since the insertion of this much reactivity would cause a power excursion, the increased fuel temperature indication to the temperature monitor and the power monitor would cause the reactor to scram. The scram would effectively shut down the reactor. There is an additional barrier to sudden positive reactivity insertions. Only $3.0 of excess reactivity is available at 250 kW in accordance with the Technical Specifications. Table 13.2 outlines the maximum fuel temperatures experience for sudden insertions of positive reactivity.

Table 13.2
Maximum centerline fuel temperatures as a function of positive reactivity insertion

Reactivity Insertion ($\rho_\$$)	Temperature (0 W)	Temperature (250 kW)
1.5	178	598
2.0	333	753
2.5	487	907
3.0	639	1059
3.5	790	1210
4.0	940	1360
4.5	1089	1509

The insertion of $3.60 reactivity is physically impossible for the control rods to achieve and the UUTR operational procedures prohibit the movement of fuel elements during reactor operation, this value of a sudden positive reactivity insertion is highly improbable.

13.3 Mishandling or Malfunction of Fuel

The fuel elements will produce radioisotopes during reactor operation. Under normal conditions, the radioisotopes will be contained within the fuel elements. To examine the highly unlikely event of a severe fuel cladding failure, a safety analysis of the release of these radioisotopes and the resulting effective dose was made. An estimate of the UUTR's fuel inventory is made IN RSAC-5 (Appendix B). This inventory assessment is based upon the reactor operating at full power for 60 hours every six months for 5 years at 1.1 MW. The assumption creates conservative estimates by a factor of 4.

13.3.1 Dose to Restricted Area

The resultant effective dose, assuming submersion is 5.83 mR /breath. Therefore an occupational dose to the worker in the reactor room for 10 minutes without respiratory protection following the cladding failure of a single fuel element in water is 1.2 R. If evacuation occurs within 2 minutes, as it no doubt will because the reactor room is small and easy to exit, the effective dose drops to 235 mR. All of these doses are within the NRC guidelines for occupational exposure as stated in 10CFR20.1203.

13.3.2 Dose to Unrestricted Area

Assuming the worst-case scenario

- all the fission products are released instantaneously out the stack (2.86 Ci or 1360 rem effective dose)
- individual is located at the centerline of the plume.
- Wind speed is 10 m/s (SLC, Utah)

The dose to that individual is 0.0624 mRem/min. The allowed dose in the unrestricted area will be reached in 26 hours. If the wind speed is reduced to 1m/s. the allowed dose in the unrestricted area will be reached in 2 hours.

13.4 Natural Phenomena

A variety of natural phenomena can affect the UUTR. These include events such as earthquakes, severe storms, floods, etc. This section will discuss the possible effects of these events on UUTR operation and safety.

13.4.1 Earthquake

Recent history indicates that the region encompassing the state of Utah and adjacent areas are zones of moderate seismic activity and minor consequence. In recent history, seismic occurrences have been low intensity events with little or no resulting damage. Should an earthquake occur of significant severity, the consequences to the UUTR facility should not cause events more severe than those discussed in preceding sections of this chapter. Since the reactor pool was constructed of a large mass of reinforced concrete, it will tend to vibrate as a single unit with a frequency of oscillation that may be different from that of the surrounding building. In addition, the TRIGA tank was built in an earthquake proof fashion with double wall and sand construction. Dislocation between the reactor and surrounding structures could occur. This could lead to a break in beam port tubes and other penetrations, which can cause the pool water to drain. This non-instantaneous loss of coolant/ shielding event was analyzed previously. An earthquake of sufficient severity to cause dislocation of the reactor pool or surrounding structures would undoubtedly be recognized and the reactor, if operating at the time, could be manually shutdown and not restarted until the integrity of the reactor pool had been established.

The most severe consequence of a major earthquake would be the failure of the reactor's aluminum tank. Because of the containment design employing a double wall, sand, and concrete construction, the only credible damage scenario would result in a slow (many hours) draining of the pool water. Again, this loss of pool water has been analyzed earlier. This condition would be easily recognizable allowing sufficient time to shut down and secure the reactor, evacuate personnel, and consider possible mitigation actions for minimizing radiation release and exposures.

Major seismic events would probably cause other related events, which would cause a reactor shutdown without operator intervention. For example, loss of building power would initiate a shim-safety rod insertion, since the control system is fail-safe with respect to power loss. Also, loss of significant quantities of pool water will cause a reactor trip from low water level in the pool.

13.4.2 Severe Storms, Floods

Occurrence of severe storms or tornadoes might cause considerable damage to the reactor building. This damage would not be caused by or even compounded by the UUTR reactor. Furthermore, the likelihood of either type of weather phenomena occurring in the Salt Lake Valley is extremely low. The depth of water above the core and thick concrete walls provide considerable protection from direct wind damage and debris impact. Should damage be sufficient to cause a breach of the pool wall, a non-instantaneous loss of coolant accident could be the result. Again, this event was analyzed previously. Ample warning of the approach of severe weather (local emergency sirens, and a weather alert radio are available) will allow appropriate actions to be taken, such as securing of the reactor and protection of personnel. Should off-site power fail, the control system (being fail-safe in the event of power loss) will cause a reactor shutdown.

Area flooding should not pose a threat to the integrity of the core or the reactor pool. The reactor is located 400 feet above the valley floor on the foothills of the Wasatch Mountains, so it is unlikely that flood water will mix with pool water. Also, the local peaks and canyons are not high enough above the snow line for spring melt to pose a credible flooding risk. The reactor site is not at or near the elevation of the nearest river flood plain. Events leading to area flooding are easily recognizable and allow ample time to shutdown and secure the reactor facility.

13.5 Man-Made Phenomena

Various man-made phenomena can result in detrimental effects upon the reactor facility. Identification and mitigation to these events are discussed in the security plan.

13.6 References

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Chapter 14

Technical Specifications

Note: changes from the current TS are highlighted, crossed-out and in red text

FACILITY LICENSE R-126

TECHNICAL SPECIFICATIONS

AND BASES

FOR THE UNIVERSITY OF UTAH

TRIGA REACTOR

DOCKET 50-407

TECHNICAL SPECIFICATIONS AND BASES FOR THE UNIVERSITY OF UTAH TRIGA NUCLEAR REACTOR

This document constitutes the Technical Specifications for the Facility License No. R-126 and supersedes all prior Technical Specifications. Included in these Technical Specifications are the "Basis" to support information the selection and significance of the specification. The bases are included for information purposes only. They are not part of the technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere. Furthermore, the dimensions, measurements and other numeric values given in these specifications may differ slightly from actual values because of normal construction and manufacturing tolerances, or normal degree of accuracy of the instrumentation.

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1.0 DEFINITIONS

The following frequently used terms are herein explicitly defined to ensure uniform interpretation of the Technical Specifications.

1.1 Reactor Operating Conditions

Abnormal Occurrence: An abnormal occurrence is defined for the purposes of reporting requirements of Section 208 of the Energy Reorganization Act of 1974 (PL 93-438) as an unscheduled incident or event which the Nuclear Regulatory Commission determines is significant from the standpoint of public health or safety.

Cold Critical: The reactor is in the cold critical condition when it is critical with the fuel and bulk water temperature both below 40°C.

Reactor Operation: Reactor operation is any condition wherein the reactor is not secured.

Reactor Secured: The reactor is secured when all the following conditions are satisfied:

(1) The reactor is shut down.

(2) The console key switch is in the "off" position and the key is removed from the console and under the control of a licensed operator or stored in a locked storage area.

(3) No work is in progress involving incore fuel handling or refueling operations, maintenance of the reactor or its control mechanisms, or insertion or withdrawal of incore experiments.

Reactor Shutdown: The reactor is shut down when the reactor is subcritical by at least $1.00 of reactivity.

Reportable Occurrence: A reportable occurrence is any of the following that occur during reactor operation:

(1) Operation with any safety system setting less conservative than specified in Section 2.2. "Limiting Safety System Settings."

(2) Operation in violation of a limiting condition for operation.

(3) Operation with a required reactor or experiment safety system component in an inoperative or failed condition which could render the system incapable of performing its intended safety function.

(4) Any unanticipated or uncontrolled change in reactivity greater than $1.00.

(5) An observed inadequacy in the implementation of either administrative or procedural controls, to such a degree that the inadequacy could have caused the existence or development of a condition which could result in operation of the reactor outside the specified safety limits.

(6) A measurable release of fission products into the environment.

Shutdown Margin: Shutdown margin shall mean the minimum shutdown reactivity necessary to provide confidence that (1) the reactor can be made subcritical by means of the control and safety systems, starting from any permissible operating conditions; and (2) the reactor will remain subcritical without further operator action.

1.2 Reactor Experiments and Irradiations

Experiment: Experiment shall mean: (1) any apparatus, device, or material which is not a normal part of the core or experimental facilities, but which is inserted into these facilities or is in line with a beam of radiation originating from the reactor; or (2) any operation designed to measure reactor parameters or characteristics.

(1) Routine Experiment. Routine experiments are those which have been previously tested in the course of the reactor program.

(2) Modified Routine Experiment. Modified routine experiments are those which have not previously been performed but are similar to routine experiments in that the hazards are neither greater nor significantly different than those for the corresponding routine experiment.

(3) Special Experiment. Special experiments are those which are not routine or modified routine experiments.

Experimental Facilities: Experimental facilities shall mean beam ports, including extension tubes with shields, thermal columns with shields, vertical tubes, incore irradiation baskets or tubes, pneumatic transfer systems, and any other inpool irradiation facilities.

Irradiation: Irradiation shall mean the insertion of any device or material that is not a normal part of the core or experimental facilities into an irradiation facility so that the device or material is exposed to a significant amount of radiation available in that irradiation facility.

Irradiation Facilities: Any inpool experimental facility that is not a normal part of the core and that is used to irradiate devices and materials.

Secured Experiment: A secured experiment shall mean any experiment that is held firmly in place by a mechanical device or by gravity, is not readily removable from the reactor, and that requires one of the following actions to permit removal:

(1) removal of mechanical fasteners

(2) use of underwater handling tools

(3) moving of shield blocks or beam port components

1.3 Reactor Components

Flux Trap: A flux trap is any region within the core whose composition is modified to change the neutron flux.

Hexagonal Ring: A hexagonal ring is one of the concentric hexagonal bands of fuel elements surrounding the central position of the core referred to as the A position. They are designated by letters starting with B for the innermost hexagonal ring.

Instrumented Fuel Rod: An instrumented fuel rod is a special fuel rod in which thermocouples have been embedded for the purpose of measuring the fuel temperatures during reactor operation.

Operational Core: An operational core is any arrangement of TRIGA fuel that is capable of operating at the maximum licensed power level and that satisfies all the requirements of the Technical Specifications.

Regulating Control Element: Regulating control element shall mean a low worth control element that may be positioned either manually or automatically by means of an electric motor-operated positioning system and that need not have a scram capability.

Seven Element Position: A hexagonal section located at the A and B hexagonal rings in the core which can be removed from the upper grid plate for insertion of specimens up to 4.4 inches in diameter after relocation of the six B-ring elements and removal of the central thimble.

Standard Control Element: Standard control element shall mean any control element that has a scram capability that is utilized to vary the reactivity of the core, and that is positioned by means of an electric motor-operated positioning system.

Standard Fuel: Standard fuel is TRIGA fuel that contains a nominal 8.5 to 20 weight percent of uranium with a U-235 enrichment of less than 20%.

1.4 Reactor Instrumentation

Channel Calibration: A channel calibration consists of comparing a measured value from the measuring channel with a corresponding known value of the parameter so that the measuring channel output can be adjusted to respond with acceptable accuracy to known values of the measured variables.

Channel Check: A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification may include comparison with independent channels measuring the same variable or other measurements of the variables.

Channel Test: A channel test is the introduction of a signal into the channel to verify that it is operable.

Experiment Safety Systems: Experiment safety systems are those systems, including their associated input circuits, that are designed to initiate a scram for the primary purpose of protecting an experiment or to provide information that requires manual protective action to be initiated.

Limiting Safety Systems Settings: Limiting safety systems settings are the settings for automatic protective devices related to those variables having significant safety functions.

Measured Value: The measured value is the magnitude of that variable as it appears on the output of a measuring channel.

Measuring Channel: A measuring channel is the combination of sensor, interconnecting cables, or lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a variable.

Operable: A system, device, or component shall be considered operable when it is capable of performing its intended functions in a normal manner.

Reactor Safety Systems: Reactor safety systems are those systems, including their associated input circuits, designed to initiate a scram for the primary purpose of protecting the reactor or to provide information that requires protective action to be initiated.

Safety Channel: A safety channel is a measuring channel in the reactor safety system.

Safety limits: Safety limits are limits on important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit – Fuel Element Temperature

Applicability: This specification applies to the maximum temperature of the reactor fuel.

Objective: The objective is to define the maximum fuel temperature that can be permitted with confidence that a fuel cladding failure will not occur.

Specifications:

(l) The temperature in a stainless-steel clad, high hydride fuel element shall not exceed 1000°C under any conditions of operation.

(2) The temperature in an aluminum clad low hydride fuel element shall not exceed 530°C under any conditions of operation.

Bases: The important parameter for a TRIGA reactor is the fuel rod temperature. This parameter is well suited as a single specification especially since it can be measured. A loss in the integrity of the fuel rod cladding could arise from a buildup of excessive pressure between the fuel moderator and the cladding, if the fuel temperature exceeds the safety limit. The pressure is caused by the presence of air, fission product gases, and hydrogen from the disassociation of the hydrogen and zirconium in the fuel moderator. The magnitude of this pressure is determined by the fuel-moderator temperature and the ratio of hydrogen to zirconium in the alloy.

The safety limit for the high hydride TRIGA fuel is based on data, including the large mass of experimental evidence obtained during high performance reactor tests on this fuel. These data indicate that the stress in the cladding because of hydrogen pressure from the disassociation of zirconium hydride will remain below the ultimate stress, provided that the temperature of the fuel does not exceed l000°C and the fuel cladding is water cooled.

The safety limit for the low hydride fuel elements is based on avoiding the phase change in the zirconium hydride which might cause excessive distortion of a fuel element. This phase change takes place at 530°C. Additional information is given in Technical Foundations of the TRIGA Report GA-471, pages 63-72, August 1958.

2.2 Limiting Safety System Settings

Applicability: This specification applies to the settings that prevent the safety limit from being reached.

Objective: The objective is to prevent the safety limits from being exceeded.

Specifications:

(1) For a core composed entirely of stainless steel clad, high hydride fuel elements or a core composed of stainless steel clad, high hydride fuel elements with low hydride fuel elements in the F or G hexagonal ring only, limiting safety system settings apply according to the location of the instrumented fuel as indicated in the following table:

Location of Instrumented Fuel Rod	Limiting Safety System Setting
B-hexagonal ring	800°C
C-hexagonal ring	755°C
D-hexagonal ring	680°C
E-hexagonal ring	580°C

(2) For a core with low hydride fuel elements installed in other than the F or G hexagonal ring, limiting safety system settings apply according to the location of the instrumented fuel rod, as indicated in the following table:

Location of Instrumented Fuel Rod	Limiting Safety System Setting
B-hexagonal ring	460°C
C-hexagonal ring	435°C
D-hexagonal ring	390°C
E-hexagonal ring	340°C

(3) For a core containing flux traps, the limiting safety system settings given in either of the above two tables must be applied to the anticipated hottest fuel element in the core.

Bases: Stainless steel clad, high hydride fuel element: The limiting safety system settings that are indicated represent values of the temperature, which if exceed, shall cause the reactor safety system to initiate a reactor scram. Since the fuel element temperature is measured by fuel elements designed for this purpose, the limiting settings are given for different locations in the fuel array. Under these conditions, it is assumed that the core is loaded so that the maximum fuel temperature is produced in the B-hexagonal ring.

The margin between the safety limit of 1000°C and the limiting safety system setting of 800°C in the B-hexagonal ring was selected to assure that conditions would not arise which would allow the fuel element temperature to approach the safety limit. The safety margin of 200°C accounts for differences between the measured peak temperature and calculated peak temperature encountered during operation and for uncertainty in temperature channel calibration. The thermocouples that measure the fuel-moderator temperature are located nominally midway between the fuel axial centerline and the fuel edge.

During steady-state operations, the equilibrium temperature is determined by the power level, the physical dimensions and properties of the fuel element, and the parameters of the coolant. Because of the interrelationship of the fuel-moderator temperature, the power level, the changes in reactivity required to increase or maintain a given power level, any unwarranted increase in the power level would result in a relatively slow increase in the fuel-moderator temperature. The margin between the maximum setting and safety limit would assure a shutdown before conditions could result that might damage the fuel elements.

Low hydride fuel element: The 460°C maximum limit for the safety system setting gives an ample margin that assures that the safety limit would not be reached through errors in measurement. Temperatures of 460°C have been shown to be safe through extensive operating experience. The surveillance requirement on the measurements of the fuel dimensions will give control over changes that result from the thermal cycling during operation.

The temperatures shown for C. D and E hexagonal ring locations were derived using the power distributions from report GA-4339.

The maximum fuel element temperatures calculated on page III-8 of the UUTR SAR 1985 were 245°C at 100 kW and 440°C at 250 kW. However experimental data based upon 363 critical reactor operations indicate a maximum fuel temperature of 114°C at 100 kW and an extrapolated maximum fuel temperature of 314°C at 300 kW.

3.0 LIMITING CONDITIONS OF OPERATION

3.1 Normal Operation

Applicability: This specification applies to the energy generated in the reactor during normal operation

Objective: The objective is to ensure that the fuel temperature safety limit will not be exceeded during normal operation

Specifications: The reactor power level shall not deliberately be raised above 250 kW under any conditions of operation

Basis: Thermal and hydraulic calculations performed by the vendor indicate that TRIGA fuel may be safely operated up to power levels of at least 2.0 MW with natural convection cooling.

3.2 Reactivity limitations

Applicability: These specifications apply to the reactivity condition of the reactor and the reactivity worth of control elements and experiments.

Objective: The objective is to ensure that the reactor can be shut down at all times and to ensure that the fuel temperature safety limit will not be exceeded.

Specifications: The reactor shall not be operated unless the following conditions exist:

(1) The shutdown margin referred to the cold-critical xenon-free condition, with the highest worth rod fully withdrawn is greater than $0.50;

(2) The rate of reactivity insertion by control rod motion shall not exceed $0.30 per second;

(3) Any experiment with a reactivity worth greater than $1.00 is securely fastened so as to prevent unplanned removal from or insertion into the reactor;

(4) The excess reactivity for the cold critical, xenon free condition is less than $2.80;

(5) The reactivity worth of an individual experiment is not more than $2.80.

Bases: The shutdown margin required by specification 3.2(1) is necessary so that the reactor can be shutdown from any operating condition and remain shutdown after cool down and xenon decay even if one control rod should remain in the fully withdrawn position.

Specification 3.2(2) assures that power increases caused by rod motion will be terminated by the reactor safety system before the fuel temperature safety limit is exceeded.

It is assumed that the worst reactivity insertion accident from unrestrained motion of the control rods is initiated from a condition corresponding to reactor startup, between 1 and 100 milliwatts, with a subcritical condition corresponding to the source level. Then 30¢/sec of reactivity insertion continues until the power level scram is tripped. A further delay of 0.1 seconds is used until the control rods begin to insert reactivity at the rate of $3/sec until the rods are inserted. Assuming no thermodynamic feedback occurs (making the calculation quite conservative) it is found that 1.7 mW-sec of energy is produced by the excursion, raising the fuel temperature by 20°F in the peak fuel flux location of the core. This temperature rise is far below the allowable rise to the fuel damage point starting from ambient conditions at startup.

Specifications 3.2(3) is based on Section 8.5 of the UUTR SAR 1985 which indicates that as much as $3.00 reactivity could be inserted in a pulse from a power level of 3 Mwt without violation of the fuel temperature safety limit. By restricting each experiment to reactivity worth of one dollar, an ample margin is provided to allow for uncertainties in the information and the uncertainty in the worth of an experiment.

Specifications 3.2(3) through 3.2(5) are intended to provide additional margins between those values of reactivity changes encountered during the course of operations involving experiments and those values of reactivity which, if exceeded, might cause a safety limit to be exceeded.

3.3 Control and Safety System

3.3.1 Scram Time

Applicability: This specification applies to the time required for the scrammable control rods to be fully inserted from the instant that a safety channel variable reaches the safety system setting.

Objective: The objective is to achieve prompt shutdown of the reactor to prevent fuel damage.

Specification: The scram time from the instant that a safety system setting is exceeded to the instant that the slowest scrammable control rod reaches its fully inserted position shall not exceed 2 seconds. For purposes of this section, the above specification shall be considered to be satisfied when the sum of the response time of the slowest responding safety channel, plus the fall time of the slowest scrammable control rod, is less than or equal to 2 seconds.

Basis: This specification ensures that the reactor will be promptly shut down when a scram signal is initiated. Experience and analysis have indicated that for the range of transients anticipated for a TRIGA reactor, the specified scram time is adequate to ensure the safety of the reactor.

3.3.2 Reactor Control System

Applicability: This specification applies to the information that must be available to the reactor operator during reactor operation.

Objective: The objective is to require that sufficient information is available to the operator to ensure safe operation of the reactor.

Specification: The reactor shall not be operated unless the measuring channels listed in the following table are operable.

Measuring Channel	Minimum Number Operable
Fuel Element Temperature	1(b)
Reactor Power Level	2
Startup Count Rate	1
Reactor Tank Water Level	1
Area Radiation Monitor	1(a)
Continuous Air Radiation Monitor	1(a)

(a) For periods of time for maintenance to the radiation monitoring systems, the intent of this specification will be satisfied if the installed systems are replaced with portable gamma-sensitive instruments having their own alarms or which shall be kept under visual observation.

(b) For periods of time for maintenance to the standard instrumented fuel element, the reactor shall be in the shutdown condition with all control rods fully inserted, and, power to the control-rod magnets and actuating solenoids has been switched off and the key removed.

Bases: The fuel temperature displayed at the control console gives continuous information on the process variable which has a specified safety limit.

The neutron detectors assure that measurements of the reactor power level are adequately covered at both low and high power ranges. The specifications on the reactor power level indication are included in this section since the power level is closely related to the fuel temperature as shown in Section 5 and Appendix III of the UUTR SAR 1985.

The reactor tank water level detector provides early information of a possible leak in the reactor cooling system or tank.

The radiation monitors provide information to operating personnel of an emergency or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the surroundings.

3.3.3 Reactor Safety System

Applicability: This specification applies to the reactor safety system channels.

Objective: The objective is to specify the minimum number of reactor safety system channels that must be operable for safe operation.

Specification: The reactor shall not be operated unless the safety channels described in the following table are operable.

Safety System Measuring Channel	Minimum Number Operable	Function
Fuel element temperature	1(a)	Scram at or below Limiting Safety System Setting
Reactor power level	2	Scram at 120 percent of full licensed power
Manual console scam button	1	Manual scram
Magnet current key switch	1	Manual scram
Console power supply	1	Scram on loss of electrical power
Reactor tank water level	1	Scram at one foot below normal operating level
Startup count rate interlock	1	Prevent control rod withdrawal when neutron count rate is less than 2 counts per second
Control rod withdrawal interlocks	All control rods	Prevent manual withdrawal of more than one control rod simultaneously

(a) For periods of time for maintenance to the standard thermocouple fuel element, the reactor shall be in the shutdown condition with all control rods fully inserted, and power to the control-rod magnets and actuating solenoid has been switched off and the key removed.

Bases: The fuel temperature scrams provide the protection to assure that, if a condition results in which the limiting safety system setting is exceeded, an immediate shutdown will occur to keep the fuel temperature below the safety limit. The power level scrams are provided as redundant protection against abnormally high fuel temperature and to assure that the reactor operation stays within the licensed limits. The equivalent operation with scrams at 120 percent of full power or **300 kW** assures that the reactor operation will terminate well below that power level above which safe cooling may not be available and also at a level below the level temperature scram trip. The manual scrams allow the operator to shut down the system if an unsafe or abnormal condition occurs. In the event of failure of the console power supply, the console power supply scram provides that operation will not continue without adequate instrumentation. The reactor tank water leak occurs in the primary system or when water level is too low (the result of any cause) for adequate radiation shielding.

3.4 Argon-41 Discharge Limit

Applicability: This specification applies to the concentration of argon-41 that may be discharged_from the TRIGA reactor facility.

Objective: To ensure that the health and safety of the public are not endangered by the discharge of argon-41 from the TRIGA reactor facility.

Specification: The concentration of argon-41 released from the facility to the environment shall not exceed 4×10^{-8} μCi/ml averaged over one year.

Basis: It is shown in Appendix B of the UUTR SAR 2005 that the release of argon-41 at the above concentration will not result in exposure in unrestricted areas in excess of the limits of 10 CFR Part 20.

3.5 Engineered Safety Feature - Ventilation System

Applicability: This specification applies to the operation of the facility ventilation system.

Objective: The objective is to ensure that the ventilation system is in operation to mitigate the consequences of the possible release of radioactive materials resulting from reactor operation.

Specification: The reactor shall not be operated unless the facility ventilation system is operable, except for periods of time not to exceed 48 hours to permit repair or testing of the ventilation system. In the event of a substantial release of airborne radioactivity within the facility, the ventilation system will be secured or operated in the dilution mode to prevent the release of a significant quantity of airborne radioactivity from the facility.

Basis: It is shown that during normal operation of the ventilation system the concentration of argon-41 in unrestricted areas is below DAC. In the event of a substantial release of fission products, the ventilation system will be secured automatically. Therefore, operation of the reactor with the ventilation system shutdown for short periods of time to make repairs insures the same degree of control of release of radioactive materials (UUTR SAR 1985 Section 8.7.5).

3.6 Limitations on Experiments

Applicability: This specification applies to experiments installed in the reactor and its experimental facilities.

Objective: The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications: The reactor shall not be operated unless the following conditions governing experiments exist.

(1) Fueled experiments are limited such that the total inventory of iodine isotopes 131 through 135 in the experiment are not greater than 10 millicuries;

(2) The quantity of known explosive materials to be irradiated is less than 25 milligrams and the pressure produced in the experiment container upon accidental detonation of the explosive has been experimentally determined to be less than the design pressure of the container; and

(3) Experiments containing materials corrosive to reactor components, compounds highly reactive with water, potentially explosive materials, or liquid fissionable materials are doubly encapsulated and able to withstand any overpressure condition deemed likely to occur.

Bases: It is shown in the UUTR SAR 1985 (Section 8.7.2) that the limits of Specification 3.6(1) prevent the dose in unrestricted areas resulting from experiment failure from exceeding 10 CFR Part 20 limits from a single accidental exposure averaged on a yearly basis. Specifications 3.6(2) and 3.6(3) are intended to reduce the likelihood of damage to reactor components resulting from experiment failure.

3.7 As Low As Reasonable Achievable (ALARA) Radioactive Effluent Releases

Applicability: This specification applies to the measures required to ensure that the radioactive effluents released from the facility are in accordance with ALARA criteria.

Objective: The objective is to limit the annual population radiation exposure owing to the operation of the TRIGA reactor to a small percentage of the normal local background exposure.

Specifications:

(1) In addition to the radiation monitoring specified in Section 5.4, an environmental radiation monitoring program shall be conducted

to measure the integrated radiation exposure in and round the environs of the facility on an annual basis.

(2) The annual radiation exposure due to reactor operation, at the closest offsite point of extended occupancy shall not, on an annual basis, exceed the average local offsite background radiation by more than 20%.

(3) In the event of a significant fission product leak from a fuel rod or a significant airborne radioactive release from a sample being irradiated, as detected by the continuous air monitor, the reactor shall be shut down until the source of the leak is located and eliminated. However, the reactor may be operated on a short-term basis as needed to assist in determining the source of the leakage.

Basis: The simplest and most reliable method of ensuring that ALARA release limits are accomplishing their objective of minimal facility-caused radiation exposure to the general public is to actually measure the integrated radiation exposure in the environment on and off the site.

3.8 Primary Coolant Conditions

Applicability: This specification applies to the quality of the primary coolant in contact with the fuel cladding.

Objectives: The objectives are (1) to minimize the possibility for corrosion of the cladding on the fuel elements, and (2) to minimize neutron activation of dissolved materials.

Specifications:

(1) Conductivity of the pool water shall be no higher than 5×10^{-6} mhos/cm

(2) the pH of the pool water shall be between 5.0 and 8.0.

Bases: A small rate of corrosion continuously occurs in a water-metal system. In order to limit this rate, and thereby extend the longevity and integrity of the fuel cladding, a water cleanup system is required. Experience with water quality control at many reactor facilities has shown that maintenance within the specified limits provides acceptable control.

By limiting the concentrations of dissolved materials in the water, the radioactivity of neutron activation products is limited. This is consistent with the ALARA principle, and tends to decrease the inventory of radionuclides in the entire coolant system, which will decrease personnel exposures during maintenance and operations.

4.0 SURVEILLANCE REQUIREMENTS

4.1 General

Applicability: This specification applies to the surveillance requirements of any system related to reactor safety.

Objective: The objective is to verify the proper operation of any system related to reactor safety.

Specification: Any additions, modifications, or maintenance to the ventilation system, the core and its associated support structure, the pool or its penetrations, the pool coolant system, the control element drive mechanism, or the reactor safety system shall be made and tested in accordance with the specifications to which the systems were originally designed and fabricated or to specifications approved by the Reactor Safety Committee. A system shall not be considered operable until after it has been successfully tested.

Basis: This specification relates to changes in reactor systems that could directly affect the safety of the reactor. As long as changes or replacements to these systems continue to meet the original design specifications, it can be assumed that they meet the presently accepted operating criteria.

4.2 Safety Limit - Fuel Element Temperature

Applicability: This specification applies to the surveillance requirements of the fuel element temperature measuring channel.

Objective: The objective is to ensure that the fuel element temperatures are properly monitored.

Specifications:

(1) Whenever a reactor scram caused by high fuel element temperature occurs, the peak indicated fuel temperature shall be examined to determine whether the fuel element temperature safety limit was exceeded.

(2) The fuel element temperature measuring channel shall be calibrated semi-annually or at an interval not to exceed 8 months by the substitution of a known signal in place of the instrumented fuel element thermocouple.

(3) a channel check of the fuel element measuring channel shall be made each time the reactor is operated by comparing the indicated instrumented fuel element temperature with previous values for the core configuration and power level.

Basis: Operational experience over the past 8 years with the TRIGA system gives assurance that the thermocouple measurements of fuel element temperature have been sufficiently reliable to ensure accurate indication of this parameter.

4.3 Limiting Conditions for Operation

4.3.1 Reactivity Requirements

Applicability: These specifications apply to the surveillance requirements for reactivity control.

Objective: The objective is to measure and verify the worth, performance and operability of those systems affecting the reactivity of the reactor.

Specifications:

(1) The reactivity worth of each control rod and the shutdown margin shall be determined annually but at intervals not to exceed 15 months.

(2) The control rods shall be visually inspected for deterioration at intervals not to exceed 2 years.

Basis: The reactivity worth of the control rods is measured to ensure the required shutdown margin is available and to provide an accurate means for determining the reactivity worths of experiments inserted in the core. Past experience with TRIGA reactors gives assurance that measurement of the reactivity worth on an annual basis is adequate to ensure no significant changes in the shutdown margin. The visual inspection of the control rods is made to evaluate corrosion and wear characteristics caused by operation in the reactor.

4.3.2 Control and Safety System

Applicability: These specifications apply to the surveillance requirements for measurements, tests, and calibrations of the control and safety systems.

Objective: The objective is to verify the performance and operability of those systems and components which are directly related to reactor safety.

Specifications:

(1) The scram time shall be measured annually but at intervals not to exceed 15 months.

(2) A channel check of each of the reactor safety system channels shall be performed before each day's operation or before each operation extending more than 1 day, except for the pool level channel which shall be tested monthly.

(3) A channel calibration shall be made of the power level monitoring channels by either nuclear or calorimetric methods annually, but at intervals not to exceed 15 months.

(4) A channel test of the temperature measuring channel shall be performed semiannually, but at intervals not to exceed 8 months.

Basis: Measurements of the scram time on an annual basis is a check not only of the scram system electronics, but also is an indication of the capability of the control rods to perform properly. The channel tests will ensure that the safety system channels are operable on a daily basis or before an extended run. The power level channel calibration will ensure that the reactor will be operated at the proper power levels.

4.3.3 Radiation Monitoring System

Applicability: This specification applies to the surveillance requirements for the area radiation monitoring equipment and the continuous air monitoring system.

Objectives: The objectives are to ensure that the radiation monitoring equipment is operating and to verify the appropriate alarm settings.

Specification: The area radiation monitoring system and the continuous air monitoring system shall be calibrated annually and shall be verified to be operable at monthly intervals.

Basis: Experience has shown that monthly verification of area radiation and air - monitoring system setpoints in conjunction with annual calibration is adequate to correct for any variation in the system caused by a change of operating characteristics over a long time span.

4.3.4 Ventilation System

Applicability: This specification applies to the reactor room ventilation system.

Objective: The objective is to assure that the ventilation system is in operation to mitigate the consequences of the possible release of radioactive materials resulting from reactor operation.

Specification: The reactor shall not be operated unless the reactor room ventilation system is in operation, establishing a negative air pressure within the reactor room, except for periods of time not to exceed 48 hours to permit repair of the system.

Basis: It is shown that during normal operation of the ventilation system the concentration of argon-41 in unrestricted areas is below DAC. In the event of a substantial release of fission products, the ventilation system will be secured automatically. Therefore, operation of the reactor with the ventilation system shutdown for short periods of time to make repairs insures the same degree of control of release of radioactive materials (UUTR SAR 1985 Section 8.7.5).

4.3.5 Experiment and Irradiation Limits

Applicability: This specification applies to the surveillance requirements for experiments installed in the reactor and its experimental facilities and for irradiations performed in the irradiation facilities.

Specifications:

(1) A new experiment shall not be installed in the reactor or its experimental facilities until a hazards analysis has been performed by the Reactor Supervisor and reviewed by the Reactor Safety Committee. Minor modifications to reviewed and approved experiments may be made at the discretion of the senior operator responsible for the operations provided that the hazards associated with the modifications have been reviewed and a determination has been made that the modifications do not create a significantly different, a new, or a greater hazard than the original approved experiment.

(2) An irradiation of a new type of device or material shall not be performed until an analysis of the irradiation has been performed and reviewed by the Reactor Supervisor.

Basis: It has been demonstrated over a number of years that experiments and irradiations reviewed by the reactor staff and the Reactor Safety Committee, as appropriate, can be conducted without endangering the safety of the reactor or exceeding the limits in the Technical Specifications.

4.4 Reactor Fuel Elements

Applicability: This specification applies to the surveillance requirements for the fuel elements.

Objective: The objective is to verify the continuing integrity of the fuel element cladding.

Specifications: All fuel elements shall be inspected visually for damage or deterioration every two years. Any fuel element which appears damaged shall be measured for length and bend. The reactor shall not be operated with damaged fuel. A fuel element shall be considered damaged and must be removed from the core if:

(1) in measuring the transverse bend, its sagitta exceeds 0.125 inches over the length of the cladding,

(2) in measuring the elongation, its length exceeds its original length by 0.250 inches,

(3) a clad defect exists as indicated by release of fission products. However, the reactor may be operated on a short-term basis as needed to assist in determining the source of the leakage.

Bases: The frequency of inspection and measurement schedule is based on the parameters most likely to affect the fuel cladding. The limit of transverse bend has been shown to result in no difficulty in disassembling the core. Analysis of the removal of heat from touching fuel elements shows that there will be no hot spots resulting in damage to the fuel caused by this touching. Experience with TRIGA reactors has shown that fuel element bowing that could result in touching has occurred without deleterious effects. The elongation limit has been specified to ensure that the cladding material will not be subjected to stresses that could cause a loss of integrity in the fuel containment and to ensure adequate coolant flow.

4.5 Primary Coolant Conditions

Applicability: This specification applies to the surveillance of primary water quality.

Objective: The objective is to ensure that water quality does not deteriorate over extended periods of time if the reactor is not operated.

Specification: The conductivity and pH of the primary coolant water shall be measured monthly and shall be as follows:

(1) conductivity < 5 x 10-6 mhos/cm.

(2) pH between 5.0 and 8.0

Bases: Section 3.8 ensures that the water quality is adequate during reactor operation. Section 4.5 ensures that water quality is not permitted to deteriorate over extended periods of time even if the reactor does not operate.

5.0 DESIGN FEATURES

5.1 Reactor Fuel

Applicability: This specification applies to the fuel elements used in the reactor core.

Objective: The objective is to ensure that the fuel elements are of such a design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications:

Standard TRIGA Fuel - Each individual unirradiated standard TRIGA fuel elements shall have the following characteristics:

a. High Hydride Fuel Element - Each high hydride fuel element shall contain uranium-zirconium hydride and be clad with 0.020 inch of 304 stainless steel. Each element shall contain a maximum of 20.0 weight percent uranium which has a maximum enrichment of less than 20 percent and 1.5 to 1.8 hydrogen atoms to 1.0 zirconium atom.

b. Low Hydride Fuel Element - Each low hydride fuel element shall contain uranium-zirconium hydride and be clad with 0.030 inch of aluminum or 0.020 inch of 304 stainless steel. Each element shall contain a maximum of 9 weight percent uranium which has a maximum enrichment of less than 20 percent and 0.9 to 1.6 hydrogen atoms to 1.0 zirconium atom.

Basis: These types of fuel elements have a long history of successful use in TRIGA reactors.

5.2 Reactor Core

Applicability: This specification applies to the configuration of fuel and incore experiments.

Objective: The objective is to ensure that provisions are made to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities will not be produced.

Specifications:

(1) The core shall be in arrangement of TRIGA uranium-zirconium hydride fuel elements positioned in the reactor grid plate.

(2) The reflector, excluding experiments and experimental facilities, shall be a combination of graphite, aluminum, and light water and heavy water.

Bases: Standard TRIGA cores have been used for years and their characters are well documented. Mixed cores of standard fuel have been tested and operated at a number of university reactors. Calculations, as well as measured performance of mixed cores have shown that such cores may be safely operated.

The core will be assembled in the reactor grid plate which is located in a pool of light water. Light water in combination with heavy water and graphite reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements.

5.3 Control Elements

Applicability: This specification applies to the control elements used in the reactor core.

Objective: The objective is to ensure that the control elements are of such a design as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications:

(1) The standard control element shall have scram capability and contain borated graphite, B_4C powder, or boron and its compounds in solid form as a poison in aluminum or stainless steel cladding.

(2) The regulation control element need not have scram capability and shall be a stainless steel element or contain the materials as specified for standard control elements.

Basis: The poison requirements for the control elements are satisfied by using neutron-absorbing borated graphite, B_4C powder, or boron and its compounds. These materials must be contained in a suitable clad material, such as aluminum or stainless steel, to ensure mechanical stability during movement and to isolate the poison from the pool water environment. Scram capabilities are provided for rapid insertion of the control element which is the primary safety feature of the reactor.

5.4 Radiation Monitoring System

Applicability: This specification describes the functions and essential components of the area radiation monitoring equipment and the system for continuously monitoring airborne radioactivity.

Objective: The objective is to describe the radiation monitoring equipment that is available to the operator to ensure safe operation of the reactor.

Specifications:

(1) Function of Area Radiation Monitor (gamma-sensitive instruments): Monitor radiation fields in key locations, alarm and readout at control console.

(2) Function of Continuous Air Radiation Monitor (beta-, gamma-sensitive detector with particulate collection capability): Monitors concentration of radioactive particulate activity and radioactive gases including Argon-41 in the building exhaust, alarm and readout at control console.

Basis: The radiation monitoring system is intended to provide information to operating personnel of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the surroundings.

5.5 Fuel storage

Applicability: This specification applies to the storage of reactor fuel at times when it is not in the reactor core.

Objective: The objective is to ensure that fuel that is being stored will not become critical and will not reach an unsafe temperature.

Specifications:

(1) All fuel elements shall be stored in a geometrical array where the k_{eff} is less than 0.8 for all conditions of moderation.

(2) Irradiated fuel elements and fueled devices shall be stored in an array, which will permit sufficient natural convection cooling by water or air, so that the fuel element or fueled device temperature will not exceed design values.

Basis: The limits imposed by the Specifications 5.5(1) and 5.5(2) are conservative and ensure safe storage of reactor fuel.

5.6 Reactor Building and Ventilation System

Applicability: This specification applies to the building that houses the reactor.

Objective: The objective is to ensure that provisions are made to restrict the amount of radioactivity released into the environment.

Specifications:

(1) The reactor shall be housed in a facility designed to restrict leakage. The minimum free volume in the facility shall be 5×10^8 cm^3.

(2) The reactor building shall be equipped with, a ventilation system designed to filter and exhaust air or other gases from the reactor building and release them from a stack at a minimum of 40 feet from ground level.

Basis: The facility is designed so that the ventilation system will normally maintain a negative pressure with respect to the atmosphere to minimize uncontrollable leakage to the environment. The free air volume within the reactor room is confined when there is an emergency shutdown of the ventilation system. Proper handling of airborne radioactive materials (in emergency situations) can be effected with a minimum of exposure to operating personnel.

5.7 Reactor Pool Water Systems

Applicability: This specification applies to the pool containing the reactor and to the cooling of the core by the pool water.

Objective: The objective is to ensure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding.

Specifications:

(1) The reactor core shall be cooled by natural convection water flow.

(2) All piping extending more than 5 ft below the surface of the pool shall have adequate provisions to prevent inadvertent siphoning of the pool.

(3) A pool level alarm shall be provided to indicate a loss of coolant if the pool level drops more than 2 ft below the normal level.

(4) The reactor shall not be operated with less than 18 ft of water above the top of the core.

Bases: This specification is based on, thermal and hydraulic calculations which show that the TRIGA core can operate in a safe manner at power levels up to 2700 kW with natural convection flow of the coolant water. Thermal and hydraulic characteristics of mixed cores are essentially the same as those for standard cores.

In the event of accidental siphoning of pool water through system pipes, the pool water level will drop no more than 5 ft from the top of the pool.

Loss of coolant alarm after 2 ft of loss requires corrective action. This alarm is observed in the reactor control room, and at the campus police station.

6.0 ADMINISTRATIVE CONTROL

6.1 Responsibility

The facility shall be under the direct control of a licensed Senior Reactor Operator (SRO) designated by the Reactor Supervisor (RS) who is also a licensed Senior Reactor Operator. The SRO shall be responsible to the RS for the overall facility operation including the safe operation and maintenance of the facility and associated equipment. The SRO shall also be responsible for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, federal and state regulations, and requirements of the Reactor Safety Committee.

6.2 Organization

(1) The reactor facility shall be an integral part of the Nuclear Engineering Laboratory of the University of Utah. The organization of the facility management and operation shall be as shown in Figure 6.1. The responsibilities and authority of each member of the operating staff shall be defined in writing.

(2) When the reactor is not secured, the minimum staff shall consist of

(a) Reactor Operator (RO) at the controls (may be the SRO or RS).

(b) Senior Reactor Operator (SRO) on call but not necessarily on site.

(c) Another person present at the facility complex who is able to carry out prescribed written instructions.

6.3 Facility Staff Qualifications

Each member of the facility staff shall meet or exceed the minimum qualifications of ANS 15.4 "Standard for the Selection and Training of Personnel for Research Reactors" for comparable positions.

6.4 Training

The Reactor Supervisor shall be responsible for the facility's Requalification Training Program and Operator Training Program.

6.5 Reactor Safety Committee

6.5.1 Function

The RSC shall function to provide an independent review and audit of the facility's activities including:

(1) reactor operations

(2) radiological safety

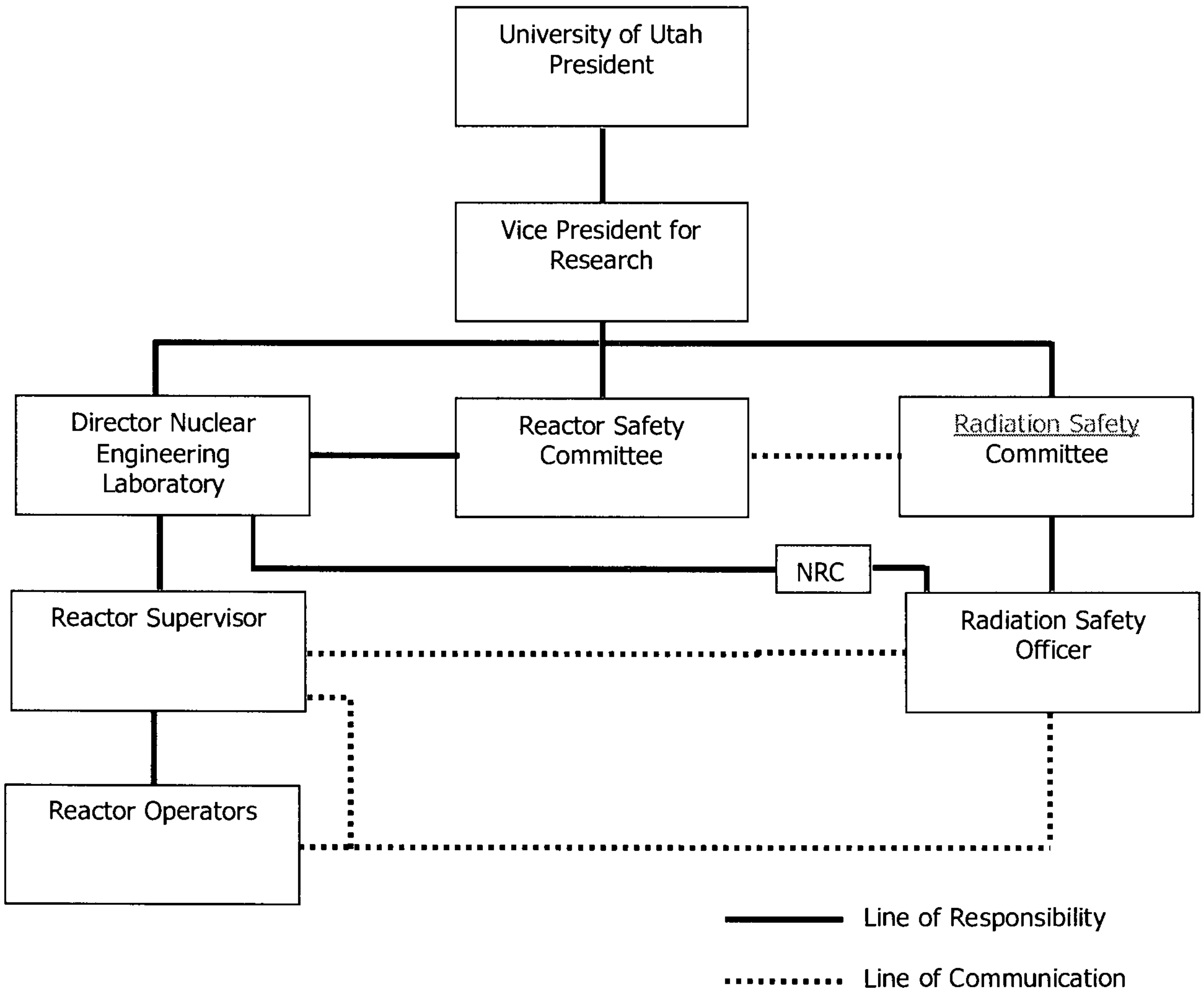

Figure 6.1

University of Utah Administrative Organization for Nuclear Reactor Operations

3) general safety

4) testing and experiments

5) licensing and reports

6) quality assurance

6.5.2 Composition and Qualifications

The RSC shall be composed of at least five members knowledgeable in fields that relate to nuclear reactor safety. The members of the committee shall include the Reactor Supervisor and faculty and staff members designated to serve on the committee. The University's Radiation Safety Officer shall be an ex officio member of the committee.

6.5.3 Operation

The Reactor Safety Committee shall operate in accordance with a written charter, including provisions for

(1) meeting frequency: the full committee shall meet at least semiannually and a subcommittee thereof shall meet at least quarterly

(2) voting rules

(3) quorums: chairman or his designate and two members

(4) method of submission and content of presentations to the committee

(5) review, approval, and dissemination of minutes

6.5.4 Reviews

The responsibilities of the RSC or designed Subcommittee thereof shall include, but is not limited to the following:

(1) review and approval of all new experiments utilizing the reactor facility

(2) review and approval of all proposed changes to the facility license by amendment. and to the Technical Specifications

(3) review of the operation and operational records of the facility

(4) review of significant operating abnormalities or deviations from normal and expected performance of facility equipment that affect nuclear safety.

(5) review of approval of all determinations of whether a proposed change, test, or experiment would constitute a change in the Technical Specifications or on unreviewed safety questions as defined by 10 CFR 50.59.

(6) review of reportable occurrences and the reports filed with the Commissions for said occurrences

(7) review and approval of all standard operating procedures and changes thereto

(8) biennial review of all standard procedures, the facility emergency plan, and the facility security plan.

6.5.5 Audits

The RSC or a subcommittee thereof shall audit reactor operations semiannually, but at intervals not to exceed 8 months. The semiannual audit shall include at least the following:

(1) review of the reactor operating records

(2) inspection of the reactor operating areas

(3) review of unusual or abnormal vents

(4) radiation exposures at the facility and adjacent environs

6.5.6 Records

The activities of the RSC shall be documented by the committee and the RSC shall maintain a file of the minutes of all meetings.

6.6 Quality Assurance

In accordance with Regulatory Guide 2.5 and ANSI 402, "Quality Assurance Program Requirements for Research Reactors," Section 2.17, the facility shall not be required to prepare quality assurance documentation for the as-built facility. Quality assurance (QA) requirements will still be limited to those specified in Section 2.17 as follows:

"All replacements, modifications, and changes to systems having a safety related function shall be subjected to a QA review. Insofar as possible, the replacement, modification, or change shall be documented as meeting the requirements of the original system or component and have equal or better performance or reliability."

"The required audit function shall be performed as specified in Section 6.5.5."

6.7 Action To Be Taken in the Event a Safety Limit is Exceeded

In the event a safety limit is exceeded:

(1) The reactor shall be shut down and reactor operation shall not be resumed until authorized by the U.S. Nuclear Regulatory Commission (NRC).

(2) An immediate report of the occurrence shall be made to the Chairman of the Reactor Safety Committee, and reports shall be made to the NRC in accordance with Section 6.10 of these specifications.

(3) A report shall be prepared that shall include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the Reactor Safety Committee for review and then submitted to the NRC when authorization is sought to resume operation of the reactor.

(4) A report shall be made to the NRC in accordance with Section 6.10 of these specifications.

6.8 Operating Procedures

Written operating procedures shall be adequate to ensure the safety of operation of the reactor, but shall not preclude the use of independent judgment and action should the situation require such. Operating procedures shall be in effect for the following items:

(1) performing irradiations and experiments

(2) startup, operation, and shutdown of the reactor

(3) emergency situations including provisions for building evacuation, earthquake, radiation emergencies, fire or explosion, personal injury, civil disorder, and bomb threat

(4) core changes and fuel movement

(5) control element removal and replacement

(6) performing preventive maintenance and calibration tests on the reactor and associated equipment

(7) power equipment

Substantive changes to the above procedures shall be made only with the approval of the Reactor Supervisor. Temporary changes to the procedures that do not change their original intent may be made by a licensed SRO. All such changes shall be documented and subsequently reviewed by the Reactor Safety Committee.

6.9 Facility Operating Records

In addition to the requirements of applicable regulations, and in no way substituting for those requirements, records and logs shall be prepared for at least the following items and retained for a period of at least 5 years for items (1) through (6) and indefinitely for items (7) through (11).

(1) normal reactor operation

(2) principal maintenance activities

(3) abnormal occurrences

(4) equipment and component surveillance activities required by the Technical Specifications

(5) experiments performed with the reactor

(6) gaseous and liquid radioactive effluents released to the environs

(7) offsite inventories and transfers

(8) fuel inventories and transfers

(9) facility radiation and contamination surveys

(10) radiation exposures for all personnel

(11) updated, corrected, and as-built drawings of the facility

6.10 Reporting Requirements

In addition to the requirements of applicable regulations, and in no way substituting for those requirements, reports shall be made to the NRC as follows:

(1) A report within 24 hours by telephone to the Project Manager, U.S.N.R.C., of

(a) any accidental release of radioactivity above permissible limits in unrestricted areas whether or not the release resulted in property damage, personal injury, or exposure;

(b) any violation of the safety limit;

(c) any reportable occurrence as defined in Section 1.1, "Reportable Occurrence," of these specifications.

(2) A report within 10 days in writing to the Document Control Center, U.S.N.R.C., Washington, D.C., with a copy to the U.S.N.R.C. Operations , of

(a) any accidental release or radioactivity above permissible limits in unrestricted areas whether or not the release resulted in property damage, personal injury, or exposure. The written report (and, to the extent possible, the preliminary telephone or telegraph report) shall describe, analyze, and evaluate safety implications, and outline the corrective measures taken or planned to prevent recurrence of the event.

(b) any violation of safety limit,

(c) any reportable occurrence as defined in Section 1.1, "Reportable Occurrence," of these specifications.

(3) A report within 30 days in writing to the Document Control Center, U.S.N.R.C., Washington, D.C. with a copy to U.S.N.R.C. Operations, of

(a) any significant variation of measured values from a corresponding predicted or previously measured value of safety connected operating characteristics occurring during operation of the reactor,

(b) any significant change in the transient or accident analysis is described in the Safety Analysis Report,

(c) any significant changes in facility organization,

(d) any observed inadequacies in the implementation of administrative or procedural controls.

(4) A report within 60 days after completion of startup testing of the reactor (in writing to the Director, Office of Nuclear Reactor Regulation, USNRC. Washington, D.C. 20555) upon receipt of a new facility license or an amendment to the license authorizing an increase in reactor power level describing the measured values of the operating conditions including:

(a) an evaluation of facility performance to date in comparison with design predictions and specifications,

(b) a reassessment of the safety analysis submitted with the license application in light of measured operating characteristics when such measurements indicate that there may be substantial variance from prior analysis.

(5) An annual report within 60 days following the 30th of June of each year (in writing) to the Document Control Center, U.S.N.R.C., Washington, D.C. with a copy to the U.S.N.R.C. Operations, providing the following information:

(a) a brief narrative summary of (i) operating experience (including experiments performed), (ii) changes in facility design, performance characteristics, and operating procedures related to reactor safety and occurring during the reporting period, and (iii) results of surveillance tests and inspections:

(b) tabulation of the energy output (in Megawatt-days) of the reactor, hours reactor was critical and the cumulative total energy output since initial criticality;

(c) the number of emergency shutdowns and inadvertent scrams, including reasons for them;

(d) discussion of the major maintenance operations performed during the period, including the effect, if any, on the safety of the operation of the reactor and the reasons for any corrective maintenance required;

(e) a brief description, including a summary of the safety evaluations of changes in the facility or in procedures and of tests and experiments carried out pursuant to 10 CFR 50.59;

(f) a summary of the nature, and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or before the point of such release or discharge:

Liquid Waste (summarized on a monthly basis)

(i) radioactivity discharged during the reporting period

- total estimated quantity of radioactivity released (in curies) an estimation of the specific activity for each detectable radionuclide present if the specific activity of the released material after dilution is greater than 1×10^{-7} μCi/cc.

- summary of the total release (in curies) of each nuclide determined just above for the reporting period based on representative isotopic analysis,

- estimated average concentration of the released radioactive material at the point of release for the reporting period in terms of μCi/cc and fraction of the applicable DAC or Effluent concentration

(ii) total volume (in gallons) of effluent water (including dilutant) released during each period of release.

Gaseous Waste (summarized on a monthly basis)

(i) radioactivity discharged during the reporting period (in curies)

- total estimated quantity of radioactivity released (in curies) determined by an appropriate sampling and counting method,

- total estimated quantity of argon-41 released (in curies) during the reporting period based on data from an appropriate monitoring system.

- estimated average atmospheric diluted concentration of argon-41 released during the reporting period in terms of μCi/cc and fraction of the applicable DAC value.

- total estimated quantity of radioactivity in particulate form with half-lives greater than 8 days (in curies) released during the reporting period as determined by an appropriate particulate monitoring system.

- average concentration of radioactive particulates with half-lives greater than 8 days released in μCi/cc during the reporting period, and

- an estimate of the average concentration of other significant radionuclides present in the gaseous waste discharge in terms of μCi/cc and fraction of the applicable DACvalue for the reporting period if the estimated release is greater than 20% of the applicable DAC.

Solid Waste (summarized on an annual basis)

(i) total amount of solid waste packaged (in cubic feet),

(ii) total activity in solid waste (in curies),

(iii) the dates of shipment and disposition (if shipped off site).

(g) An annual summary of the radiation exposure received by facility personnel and visitors in terms of the average radiation exposure per individual and greater exposure per individual and greater exposure per individual in the two groups. Each significant exposure in excess of the limits of 10 CFR 20 should be reported, including the time and data of the exposure as well as the circumstances that led up to the exposure.

(h) An annual summary of the radiation levels of contamination observed during routine surveys performed at the facility in terms of the average and highest levels.

(i) An annual summary of any environmental surveys performed outside the facility.

Chapter 15

Financial Qualifications

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15 Financial Qualifications

This chapter of the SAR presents the financial information necessary to qualify the University of Utah CENTER for owning, operating, and decommissioning the University of Utah TRIGA Reactor (UUTR), pursuant to 10 CFR 50.33(f) and (k). The information included in this chapter is non-proprietary. The University of Utah 2004 financial report is included in the relicensing package (Appendix E).

15.1 Financial Ability To Construct a Non-Power Reactor

The UUTR shall remain in "as-built" condition. No construction is necessary to fulfill the intent of the requested licensing action.

15.2 Financial Ability To Operate a Non-Power Reactor

This section discusses the financial ability of the University of Utah CENTER to operate the UUTR. The current cost to operate the reactor is approximately $250,000/year. The majority of the cost is salary and benefits for the Reactor administrator and Reactor supervisor. The University of Utah covers these salaries and associated fringe benefits. All activates other than those required for regulatory compliance are covered by the research or service contract for which the work is preformed. Additional expenses in the next four years are not anticipated at this time; however, we project a conservative increase in the cost of the same expenditures at a rate of 3% per year. The University of Utah covers the cost of insurance for the UUTR. Coverage is provided by American Nuclear Insurer's for an annual premium of approximately $8,000. Overhead costs such as utilities, confinement building maintenance and health physics monitoring are provided by the University of Utah and are excluded from this analysis of operating cost.

Chapter 16

Other License Considerations

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16 OTHER LICENSE CONSIDERATIONS

16.1 Prior Use Reactor Components

An additional purpose of this SAR is to increase the maximum operating power level of the UUTR from 100 kW(th) to 250 kW(th). This chapter of the SAR addresses safety considerations involving continued use of present reactor components. All components from the 100 kW system will be employed for the 250 kW system. The following reactor components that will be addressed in this section:

Reactor Tank
Core Assembly
Fuel and Instrumented Fuel
Neutron Source
Control Rods and Control Rod Drives
Ion Chambers
Fission Chamber
Area Radiation Monitors
Continuous Air Monitors
Reactor Console
Ventilation System
Reactor Cooling System

These components will be evaluated using the following criteria, and compared where applicable to design life criteria. Deterioration due to:

Radiation
Temperature effects: Thermal Cycling, and High Temperatures
Corrosion
Erosion
Mechanical Damage

Reactor Tank:

The aluminum reactor tank has sufficient distance from the core so that radiation damage effects are negligible. The neutron influence "nvt" threshold for significant damage to aluminum is much greater than any possible operating level of the UUTR over its lifetime. Water temperatures remain between 15 and 40 °C with very long cycling periods (10 to 12 hours), thus temperature cycling effects are negligible. The entire interior structure of the UUTR interior tank walls are clearly visible from the top of the tank and there is no corrosion or structural damage evident to date after 3 decades of operation. Additionally, there are no sources or evidence of erosion or mechanical damage to the tank structure. Therefore, in its present condition, the UUTR reactor tank is deemed acceptable for continued use at an upgraded power of 250 kW(th).

Core Assembly:

The core assembly consists of the upper, and lower grid plates, support assembly, and the side plates. These plates have been subjected to a high radiation environment and there has been no evidence of deformation or discoloration due to this exposure, and no expectation of damage at proposed power levels. Thermal cycling of these components is on the order of 15 to 100 °C with a typical period of one hour. The rise time to these sorts of temperatures is typically much faster than the cycling period. Nevertheless these temperature effects are incapable of causing any damaging or permanent deformations. A 250 kW up-grade will not create conditions that will effect the designed performance of the core assembly.

Fuel and Instrumented Fuel:

Because the surveillance procedures and design criteria for fuel and instrumented fuel are identical, they will be evaluated together. TRIGA fuel used in the UUTR is designed for a core lifetime of 7000 MW days. The since the reactor's operation began in 1974 the core has seen 17,000 kW hours (0.71 MW days) of operation or 0.01% of the fuel's operational lifetime. However, the majority of the fuel in this core comes from the University Arizona and General Atomic (GA) and had an operational use of approximately 210 MW days. As can be seen from these values, the UUTR current fuel life is well below the limits established by General Atomic. Additionally, the fuel's temperature limits (design basis limits of 530 °C for aluminum fuel, 1150 °C for stainless steel fuel) have never been approached (maximum operating fuel temperature of the UUTR reactor, at 100 kW, is less than 130 °C.) Furthermore, operation at 250 kW will maintain fuel temperature well below the fuel temperature limits, so the reactor's thermal limits will never be reached. To insure that corrosion, erosion, or mechanical damage to the fuel is well within acceptable limits surveillance of the fuel is performed monthly through water analysis, for fuel leaks, and every two years visual inspections of the fuel are performed to look for evidence of corrosion, erosion or mechanical damage. Thus the fuel currently used in the UUTR is acceptable for use in the 250 kW uprated core.

Neutron Source:

The UUTR neutron source is a double encapsulated Pu-Be neutron source. This source is removed from its position in the reactor core at power levels greater than 1.0 W, and therefore is not exposed to significant neutron flux at power greater than 1.0 W or the heated cooling water thermal environment. Also, by visual inspection no evidence of erosion, corrosion, or mechanical damage has been observed on the neutron source. Therefore, the neutron source is acceptable for continued use at an upgraded power of 250 kW

Control Rods and Control Rod Drives:

Because of the routine surveillance procedures for the control rods and the integrally connected control rod drives , these systems will be evaluated together. The control rods and control rod drives are physically inspected for erosion, corrosion, and mechanical damage every two years. Every six months the rods and rod drives are calibrated and the rod reactivity worths are determined experimentally. These detailed and documented surveillance procedures employed for future operation at 250 kW will ensure that the control rods are free of radiation contamination, corrosion, erosion, and mechanical damage after the upgrade. The control rods and associated systems are considered acceptable for use at an upgraded power of 250 kW.

Ion Chambers:

The three ion chambers used to measure linear, log, and percent power are calibrated every six months via a thermal power calibration procedure. If problems develop with one of these chambers, the chamber will be removed from service, and repaired or replaced as a necessary maintenance procedure for the reactor. As part of the restart at 250 kW the non-linearity of the chambers response will be addressed. The UUTR will not be operated without a full complement of nuclear instrumentation required in the Technical Specifications. Maintaining these surveillance procedures after the 250 kW upgrade will continue to ensure that the ion chambers are free of radiation, corrosion, erosion, and mechanical damage after the upgrade. The present ion chambers are deemed acceptable for use at an upgraded power of 250 kW.

Fission Chamber:

The fission chamber is used to verify a continuous neutron population in the UUTR core. If a problem develops with the fission chamber, it will be removed from service, and repaired or replaced as a maintenance procedure for the reactor. Maintaining these surveillance procedures after the 250 kW upgrade will continue to ensure that the fission chamber is free of radiation, corrosion, erosion, and mechanical damage after the upgrade. Therefore, the present fission chamber is deemed acceptable for continued use at an upgraded power of 250 kW.

Area Radiation Monitors:

The function, performance, and use of the ARM's does not change under an upgrade to 250 kW. Therefore, the present ARM system is acceptable for continued use at an upgraded power of 250 kW.

Continuous Air Monitors:

The function, performance, and use of the CAM does not change under an upgrade to 250 kW. Therefore, the present CAM system is acceptable for continued use at an upgraded power of 250 kW.

Reactor Console:

The current reactor console is designed to meet all Technical and operation specifications required for 250 kW operation. Furthermore, the function, performance, and use of the console does not change under an upgrade to 250 kW. Therefore, the present console is acceptable for continued use at an upgraded power of 250 kW MW.

Ventilation System:

The current ventilation system is designed, as an engineered safety system, to meet all Technical and operation specifications of 250 kW operation. Furthermore, the function, performance, and use of the ventilation system does not change under an upgrade to 250 kW. Therefore, the present ventilation system is acceptable for continued use at an upgraded power of 250 kW.

Reactor Cooling System:

The current reactor cooling system is designed to reject 25 kW of thermal power. As result the allowable reactor run time will be pool water temperature limited. While this cooling system cannot provide full heat removal for continuous 250 kW operation, the function, performance, and use of the cooling system does not change under an upgrade to 250 kW. The cooling system is acceptable for pre-cooling the reactor water to extend the length of higher power level runs. Therefore, the present reactor cooling system is acceptable for continued use at an upgraded power of 250 kW.

16.2 Medical Use of Non-Power Reactors

No human-use irradiation studies will be performed at the UUTR. The University of Utah's Radiation Safety Committee's Human Use Subcommittee must review and approve through appropriate designated medical personnel all such applications. Furthermore, while the UUTR may conduct research on the production of medical isotopes there are no plans to produce any medical isotopes intended for human use. Approval for such use must be obtained from the above medical channels as well as both Utah and NRC regulatory agencies. If either of these circumstances change the appropriate analysis will be performed, and the application for facility amendment will be sent to the NRC.

Chapter 17

Decommissioning And Possession-Only License Amendments

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17 Decommissioning and Possession-Only License Amendments

At this time the University of Utah is not considering decommissioning of the UUTR. Therefore a preliminary decommissioning plan is not included nor required in this SAR. If and when the UUTR is considered for decommissioning, the following actions will be taken in sequence:

1) Five years prior to termination of license, a preliminary decommissioning plan will be submitted to the NRC.
2) Next, a detailed decommissioning plan will be submitted to the NRC.
3) Following these activities, an application for a termination of license will be submitted to the NRC.

During this period of decommissioning an application for a possession-only license may be made after the reactor fuel has been shipped off site. All decommissioning plans, amendments, and applications to license will be made in accordance with 10 CFR 50.82, 10 CFR 50.75(f), and Draft NUREG/CR-5849.

Environmental Report

University of Utah

Center for Excellence in Nuclear Technology, Engineering, and Research

License: R-126

Docket: 50-407

January 7, 2005

1. Introduction

This environmental report is prepared in accordance with 10 CFR 51 to support the nuclear reactor license renewal application at the University of Utah. The University of Utah TRIGA Reactor (UUTR) is a light-water cooled and moderated reactor using uranium fuel enriched at <20%. The reactor is currently licensed to operate at thermal power levels up to 100 kW and application has been made to increase the maximum thermal operating power to 250 kW. The reactor is housed in the Merrill Engineering Building on the main campus of the University of Utah. A full description of the reactor is contained in the University of Utah, Safety Analysis Report, License R-126 Docket Number 50-407. The reactor and the associated laboratories, which support the reactor, are an essential teaching and research component of the Nuclear Engineering Program at the University of Utah.

2. Proposed Actions

We propose to continue operating the UUTR that was initially licensed in 1975. The UUTR license was renewed in April 1985 to extend to April 2005. The UUTR has a history of safe and reliable operations. With the application to extend the operating license beyond the initial 30-year authorization, an increase the maximum steady-state thermal power to 250 kW is also being requested.

3. Impact of the Proposed Action

The Center for Excellence in Nuclear Technology, Engineering, and Research (CENTER) is founded on the following premise: Education through Research and Service. The CENTER is the research entity on campus that is responsible for the UUTR operations and administration. The UUTR is operated solely for educational and research purposes. The reactor significantly contributes to the local and larger regional and national education community. The impact the proposed action of extending the UUTR operating license on each component education, research and service is identified in the following sections.

3.1 Education

The reactor is the keystone of the Nuclear Engineering Program (NEP) at the University of Utah. The UUTR is operated and maintained by the faculty and students of the NEP and provides comprehensive experience in research and training reactor operations and safety. The NEP courses include imbedded laboratories focused on reactor physics, operations, safety, regulations, radiation sciences, and other applications. The NEP typically graduates 2 Masters and 2 PhDs Nuclear Engineering Degrees and associated minors annually. In addition the NEP curriculum is accepted as undergraduate technical electives in Civil and Environmental Engineering, Chemical Engineering, Mechanical Engineering, and Electrical Engineering. This cross listing of the Nuclear Engineering curriculum promotes greater understanding and diversity for students in these disciplines. The past success of these nuclear technical electives has stimulated the NEP to create new coursework and begin the process of establishing a minor in Nuclear Engineering at the Bachelor of Science level. The faculties that service the NEP curriculum are multidisciplinary with appointments in Civil and

Environmental Engineering, Chemical Engineering, Mechanical Engineering, and the School of Medicine.

3.2 Research

The UUTR and the associated Radiochemistry Laboratories support a variety of research programs including radiochemistry, fission track analysis for plutonium detection, neutron induced autoradiography for determining distribution of plutonium in samples, Monte Carlo N-Particle (MCNP) modeling for reactor physics and dose reconstruction, and neutron activation analysis. These NEP research programs have crossed international boundaries including agreements with the Russian, Australian, and Marshall Island governments. National laboratories, other universities, federal contractors, and private industry have employed the graduate students funded by these research programs. The CENTER faculty, staff, and students are committed to expanding these research programs, publishing significant results, and improving nuclear engineering and nuclear science education and training. The NEP at the University of Utah is an important contributor to the scientific and technical manpower needs for the US.

3.3 Service

The UUTR also provides service (e.g., radiation services, health physics, regulatory support, etc.) and support for local government agencies and industries. The UUTR is the only research reactor in the intermountain region and does not compete with others to provide services available commercially. The UUTR is a major educational resource for the region and provides training, tours, and other educational activities to regional universities and high schools. The UUTR is the only NIST traceable research reactor to provide "1 MeV-Silicon equivalent neutron doses". This specialized service is essential for electronic hardening and materials damage testing required by the US Air Force and supports the mission of Hill Air Force Base and Logistic Center. Other UUTR services provided to the community include outreach programs for K-12 students including tours and simple radiation experiments, workshops for science teachers ("teach the teachers") and allowing Boys and Girls Scouts to obtain "atomic energy" badges.

3.4 Future Educational Needs and Goals

The US and the world are highly dependent on nuclear technologies ranging from the smoke detectors to the sterilization sources used for medical devices. These technologies need trained engineers to create and execute new applications of radiation properties. In addition to industrial needs, nuclear power remains a vital solution to the energy needs of world. To meet these needs US universities need to maintain a viable educational infrastructure for the next generation of engineers and scientists. The University of Utah remains committed to providing a strong nuclear education through the use of the UUTR. The UUTR is operated and maintained by students, creating a true hands-on learning environment superior to programs where modeling and simulation have become the standard focus.

The NEP is currently expanding its role in the College of Engineering. Long-term planning includes relicensing the TRIGA Reactor, creating three new undergraduate courses (viz., Health Physics, Nuclear Power, and a general education class entitled "Nuclear Engineering Fact, Fiction, Responsibility and Promise"). These new courses will be used to create a minor in Nuclear Engineering at the BS level while maintaining the high standards of the graduate program.

4. Unavoidable Environmental Risks

Great effort is taken by the CENTER the operation of a reactor to minimize the generation of hazardous and radioactive materials. The UUTR as do other university research reactors generates the following low-level environmental impacts: Solid, liquid and gaseous radioactive material releases and waste heat. All releases from the UUTR are documented and reported to the Nuclear Regulatory Commission in "The University of Utah TRIGA Reactor Annual Operating Report" submitted annually each August. Table 4 is a summary of the solid, liquid and gaseous releases from the UUTR during the past 5 years. More detailed information is contained the section addressing each release category. All releases were very low and well within regulatory limits.

Table 4. Summary of UUTR Releases for the past 5 years.

Year	Release		
	Solid	Liquid	Ar-41
1999-2000	0	15.89 mCi H^3	221.72 μCi
2000-2001	2.08484 μCi	0	44.784 μCi
2001-2002	0	0	39.497 μCi
2002-2003	0	0	16.731 μCi
2003-2004	0	0	20.821 μCi

4.1 Solid Radioactive Waste

Solid radioactive wastes created in using the reactor include gloves, paper towels, spent samples and sources, resin from the demineralizer system and contaminated laboratory supplies, (glass ware, spatula's). Solid waste is transferred to the University of Utah Radiological Health Department for disposal under its Broad Form License. Table 4.1 contains a description of the specific isotopes released in 2000-2001.

Table 4-1. Solid waste shipped between 7-1-00 and 6-30-01

Isotopes	Amount (μCi)
Eu-152	0.4338
Cs-134	0.5396
Mo-99	0.0017
Co-60	0.0265
Mn-54	0.0040
Sb-124	0.6288
Nd-147	0.4406
Sc-46	0.0098

The requested power upgrade for the UUTR will not significantly affect the anticipated solid radioactive waste generated at the UUTR.

4.1.1 Fuel

The UUTR uses low enriched (<20%) TRIGA fuel. The average annual burn-up rate of the fuel is less than 1 gram U^{235} annually. Because of this low burn-up rate, no special requirement for fuel replacement is required. The U.S. Department of Energy retains ownership of all UUTR fuel. The UUTR is in the process of replacing the aluminum-clad fuel with stainless steel clad fuel. All fuel shipping will be performed in accordance with the current licensing and regulatory requirements.

4.2 Liquid Radioactive Waste

Water from routine UUTR maintenance operations is released to the sanitary sewage system. Prior to release, radiological analyses must confirm that the discharged water effluent is within regulatory release limits. This release is supervised and monitored by the Radiological Health Department. When liquid effluents may exceed regulatory release limits, these effluents are transferred to the Radiological Health Department for proper processing and disposal. In 1999 approximately 10 gallons of liquid effluent containing 0.9048 mCi of tritium was transferred to Radiological Health for disposal. This Department handles the disposal of all radioactive wastes for the University campus, research and medical facilities.

4.3 Radioactive Gases

The production and release of radioactive gases is related to the power level and run time of the UUTR. At 90 kW operations, argon-41 production is substantially below effluent concentration limits for unrestricted areas. The peak minimum detectable concentration of Ar-41 for the stack monitor is one-third of the 10 CFR 20 limit for release to unrestricted areas. The maximum release of Ar-41 from UUTR operations has been less than 10% of maximum permissible concentration (MPC) for unrestricted release. At 250 kW the release of Ar-41 will remain below the derived air concentrations and effluent concentrations defined in 10 CFR 20.

4.4 Waste Heat

The UUTR core is cooled by a natural convection. A 25kW shell and tube heat exchanger is installed to provide cooling. The heat exchanger is charged with R-134A. Heat from the R134A is transferred to an independent water source that is released to the sanitary sewer system. The heat exchanger requires a low flow rate 15gpm and the maximum temperature of the released water is less than 95 C. This release water is monitored for radiation levels. The released sewage has a negligible heating impact upon the environment.

4.5 Personnel Monitoring

The University of Utah Radiological Health Department issues and monitors personnel with duties in the reactor laboratory on a regular or an occasional basis as necessary. Radiological Health Department provides radiation training and dosimetry for all University personnel accessing radioactive materials or radiation areas as necessary. The duty category and monitoring period for personnel at the UUTR for the past 5 years are summarized in Table 4.5

Table 4.5 CENTER Reactor Personnel Exposure

Year	Numbers of monitored persons in annual–dose category		
	< 0.1 rem immeasurable	0.1 –0.5 rem	> 0.5 rem
1999-2000	15	0	0
2000-2001	17	0	0
2001-2002	13	0	0
2002-2003	13	0	0
2003-2004	12	0	0

4.6 Environmental Monitoring

Six environmental monitoring stations were established in July of 1987 for recording and documenting radiation exposures from airborne radioactivity and deposition of contamination on surrounding surfaces. Three of these stations are located on the roof of Merrill Engineering Building, where the reactor is housed. The other three stations are located on the roofs of adjacent buildings; viz., Kennecott Research Center, EMRL, and Building #80. One environmental TLD badge is placed at each station and exchanged quarterly. The data collected from 1997 to 2003 is presented in Table 4.6. The University of Utah Radiological Health Department administers these environmental stations.

5. Alternatives to Continued Operations of the UUTR

Each US university research reactor is a unique facility, with individual educational and research objectives. The loss of any of the remaining US university reactors would constitute a significant weakening of the US ability to operate and control nuclear related facilities. A National Academy Study performed recently confirmed this observation. The UUTR is unique and an essential training and research tool. Students and faculty have direct access to the facility and intimate involvement with operations, control, and regulation. The faculties of the NEP are fully committed to maintaining close ties between the reactor and the University's educational program. However, the continued operation of UUTR is not guaranteed and is subject to changes in U.S. policy, regulatory issues, and societal pressures. The effect of the decommissioning the UUTR would probably entail significant impact and possible closure of the Nuclear Engineering Program (NEP) and loss of research and service programs. At this time there are no plans for decommissioning the UUTR by the University of Utah Administration.

Table 4.6 CENTER Environmental Monitoring Results (mRem)

Year	QTR	1	2	3	4	5	6
1997	1	28	34	35	42	35	40
	2	30	23	29	34	27	28
	3	26	22	26	34	23	33
	4	28	21	25	30	24	28
	Average quarterly readings 29±6 Background 20% = 34 Background at location #5 28±6						
1998	1	29	29	29	41	29	36
	2	28	25	29	30	23	34
	3	22.4	22.4	21.6	29	21.4	Absent
	4	21.5	23.6	26.8	34	26.2	57***
	Average quarterly readings 28±5 Bkg 20% = 34 Bkg at location #5 28±6						
1999	1	25	24.2	Absent	Absent	29	33.8
	2	25.8	29.2	26.8	32.6	29.2	34.4
	3	34.4	29.8	33.4	36.4	39.6	38.4
	4	29	33	30.6	36	41.4	38.6
	Average quarterly readings 31.2±7.9 Bkg 20% = 37.4 Bkg at location #5 34.8±6.6						
2000	1	29	33	30.6	36	41.4	38.6
	2	27.2	29.2	29.4	47	59	39.6
	3	Absent	25.2	23.2	30.2	35.2	38.6
	4	34.8	29	30.6	38.2	42.4	33.4
	Average quarterly readings 34.8±7.9 Bkg 20% = 41.8 Bkg at location #5 44.5±10.2						
2001	1	15	15	15.8	18.8	20.3	20.3
	2	26	29	29	31	37	36
	3	24	26	23	39	34	34
	4	21	24	27	28	33	31
	Average quarterly readings 26.5±7.0 Bkg 20% = 31.8 Bkg at location #5 31.1±7.4						
2002	1	33	29	32	37	39	55***
	2	25	26	26	33	36	33
	3	31	27	26	34	38	37
	4	37	35	35	42	50	42
	Average quarterly readings 34.04±6.1 Bkg 20% = 40.8 Bkg at location #5 40.75±6.29						
2003	1	35	31	32	39	42	40
	2	31	28	27	32	43	38
	3	36	35	31	39	46	42
	4	27	32	35	37	44	42
	Average quarterly readings 36±5.56 Bkg 20% = 43.2 Bkg at location #5 43.75±6.29						

***No control available for this measurement

Environmental monitor locations are defined as follows

- #1 MEB, NW - NW corner of MEB roof on steel beam
- #2 MEB, NE - NE corner of MEB roof on steel beam floor
- #3 MEB, SE - SE corner of MEB roof on steel beam
- #4 EMRL, - N end on roof facing MEB
- #5 BLDG 80 - E Window outside 2nd
- #6 KRC, N - N end on roof facing MEB

6. Analysis

The UUTR is an integral part of the University of Utah's Nuclear Engineering Program. Based on the data presented here the facility is operating with minimal radiation exposures and releases, well within regulatory limits. Personnel, environmental and Area radiation monitoring confirm that all exposures are within ALARA expectations. The UUTR is an existing facility. No capital funds are required for continued operation. The desirable and anticipated decision is that the UUTR Reactor License be renewed and the power upgrade be approved.

7. Long Term Effects on the Environment

At the eventual closure of UUTR operations, all areas housing or impacted by the UUTR and the affiliated laboratories will be decommissioned and returned to general university use. The reactor fuel (owned by the DOE) will be shipped to a designated DOE facility. Upgrade of the UUTR power level will not materially impact this outcome. Indeed, it is anticipated that the increase in power level will significantly enhance the services and research potential of the facility. The environmental impact associated with renewing the UUTR license and upgrading the power are deemed to be insignificant compared to the positive benefits resulting from enhanced educational and research opportunities offered by the University of Utah to the nation.

Safety Analysis Report

University of Utah

Center for Excellence in Nuclear Technology, Engineering, and Research

License: R-126

Docket: 50-407

March 2005

APPENDIX A- TABLE OF CONTENTS

A.1 Tank Stress Calculations

The maximum stress on the reactor for both the inner and outer tank walls occurs by hoop stress at the bottom of the reactor tank. The equation is:

$$\sigma = \frac{P \cdot r}{t}$$

where P is given by:

$$P = \Delta h \cdot g \cdot \rho$$

where:

σ = stress (Pa)
P = pressure (Pa)
r = tank radius (1.168 m, 1.829 m)
t = tank wall thickness (aluminum 0.0079375 m, steel 0.0047625 m)
Δh = water, or sand depth (7.30 m)
g = gravity (9.81 m/s^2)
ρ = density of water or sand (water 1000 kg/m^3, sand 1515 kg/m^3)

These calculations yield the following results. Stress in the inner (aluminum) tank is 10.54 MPa and the outer (steel) tanks stress is 41.67 MPa. The yield stresses are

AL – 13×10^6 psi or 8.96×10^4 MPA

STEEL – 30×10^6 psi or 2.07×10^4

Thus, these stresses are far below the yield stress values. It should be noted that these tanks are submerged at least 12 feet into the earth, and the stress relieving nature of the back fill is not taken advantage of in these calculations.

A.2 Burnup of Uranium 235

Uranium 235 burn up is accounted for in high flux, average flux, and low flux regions of the core. Burnup is described by the following:

$$\frac{dN}{dt} = -N \cdot \sigma \cdot \phi$$

which has a solution of the form:

$$N = N_o \cdot e^{-\sigma \cdot \phi\, t}$$

where:

N = atoms of U-235
t = irradiation time (1 day = $86.4 \cdot 10^3$ s)
σ = neutron capture cross section ($681 \cdot 10^{-28}$ m^2)
ϕ = neutron flux (high $5.2 \cdot 10^{16}$, average $4.14 \cdot 10^{16}$, and low $2.26 \cdot 10^{16}$ $nt/m^2 \cdot s$)
$1-N/N_o$ = fraction of burnup of U-235

These calculations have the following results. Percent burnup per mega watt day is:

High	0.031 % / MW·day
Average	0.024 % / MW·day
Low	0.013 % / MW·day

Thus for operations of 10 hours a month the fuel life time is on the order of 20 years before the 100 MW·day lifetime of the fuel is reached.

A.3 Plutonium239 Formation

During the course of operations of the UUTR Pu-239 is formed from the absorption of a neutron by U-238 and its subsequent decay chain. Pu-239 formation is described by the capture of thermal and resonance neutrons. The relation given below describes thermal neutron capture:

$$N_{239} = \phi \cdot N_{238} \cdot \sigma_{238}$$

For resonance capture the relationship is given by the number of fast neutrons from all species that are captured in resonance.

$$N_{239} = \phi \cdot \varepsilon \cdot P_l \cdot (1 - p) \cdot \sum (N \cdot \sigma_a \cdot \eta)$$

So combining the relations we get:

$$N_{239} = \phi \cdot \left\{ N_{238} \cdot \sigma_{238} + \left[\varepsilon \cdot P_l \cdot (1 - p) \cdot \sum (N \cdot \sigma_a \cdot \eta) \right] \right\}$$

where

N_{239} = atoms formed of Pu-239 per fuel rod (atoms/sec)
N_{238} = atoms of U-238 per fuel rod ($397.8 \cdot 10^{21}$ atoms)
σ_{238} = neutron capture cross section ($2.70 \cdot 10^{-28}$ m^2)
ϕ = neutron flux (average $4.14 \cdot 10^{16}$ nt/$m^2 \cdot$s)
ε = fast fission factor (1.052)
P_l = non leakage probability (0.7454)
p = resonance escape probability (0.877)
N = atoms of U-235 per fuel rod ($100.7 \cdot 10^{21}$ atoms)
σ_a = neutron capture cross section ($681 \cdot 10^{-28}$ m^2)
η = fast neutrons produced per neutron absorbed in fissile material (2.07)

[redacted]

A.4 Burn-up In Control Rods

The boron carbide control rods burn up is accounted for in the following calculations:

$$\frac{dN}{dt} = -N \cdot \sigma \cdot \phi$$

which has a solution in the form:

$$N = N_o \cdot e^{-\sigma \cdot \phi\, t}$$

where:

$$N_o = \frac{\rho \cdot N_A}{A}$$

here:

N = atoms of B
ρ = density (2250 kg/m^3)
N_A = Avogadro's number ($6.02 \cdot 10^{23}$ atoms per mol)
A = atomic number (56/4·1000 kg/mol)
t = irradiation time (1 day = $86.4 \cdot 10^3$ s)
σ = neutron capture cross section ($764 \cdot 10^{-28}$ m^2)
ϕ = neutron flux ($4.136 \cdot 10^{16}$ nt/m^2·s)
$1\text{-}N/N_o$ = fraction of burn up of B

These calculations have the following results. Percent burnup per mega watt day is 0.03 % / MW·day. Thus at 250 kW 10 % of the B_4C is removed after 400 days of 24 hour - operation (480 hours per year for 20 years). These burnup calculations are conservative based on the assumption that the rods are in the core during operation. In fact, the safety rod is fully removed and the control rods are partially removed for the core during operation.

A.5 Neutron Flux Calculations

Neutron flux calculations for the UUTR are developed from experimental data from the 100 kW core configurations. The emphasis of the 250 kW core is on a compact core with as large as possible peak flux. To accomplish the upgrade in power the aluminum and used fuel received from GA (original core) will be removed from the core. The current core configuration has enough reactivity to operate at 250 kW. Therefore experimental data from configuration "core 24b" will be used to estimate the flux profile at 250 kW. The procedure used to extrapolate the neutron flux begins with experimentally determining the thermal and total neutron flux's in the central irradiator of the core by gold and cadmium foils. Copper wires placed across the top of the core and axially in the central irradiator are used to determine the flux profiles. The flux profiles from these copper rods are normalized with respect to the gold and cadmium foils flux results, thus giving actual radial and axial flux measurements and distributions for the core.

The thermal and total fluxes were determined from irradiated gold and cadmium foils (Cd ratio). These estimations are made by the following equations. The center flux is found from the following relation.

$$\phi_{th} = \frac{CR \cdot A \cdot e^{\lambda \cdot t_{decay}}}{\varepsilon \cdot B \cdot \lambda \cdot \sigma \cdot \xi \cdot m \cdot N_A \cdot t_{irr}} \left(\frac{R_{cd} - 1}{R_{cd}} \right)$$

where

ϕ = thermal flux (nt/cm^2·s)
CR = Sample count rate (1129.675 counts/sec)
A = Atomic weight (196.967 g/mol)
λ = decay constant ($2.97848 \cdot 10^{-6}$ 1/s)
ε = detector efficiency including solid angle (0.0001827)
B = Branching ratio (.986)
σ = microscopic absorption cross section for Au-197 ($98.7 \cdot 10^{-24}$ cm^2)
ξ = sample purity factor (1.0)
m = mass (0.112 g)
N_A = Avogadro's number ($6.02 \cdot 10^{23}$ atoms/mol)
R_{cd} = Cadmium ratio (3.834)
t_{decay} = Sample decay time (1224600 s)
t_{irr} = Sample irradiation time (600 s)

The resulting thermal neutron flux was found to be $2.95 \cdot 10^{12}$ nt/cm^2·s, with a total flux of $3.98 \cdot 10^{12}$ nt/cm^2·s at 90.0 kW. Extrapolating to a power of 250 kW the new thermal $8.18 \cdot 10^{12}$ nt/cm^2·s, and the total neutron flux is $1.12 \cdot 10^{13}$ nt/cm^2. The flux's found by the gold foil method were obtained in the fuel's midpoint in the central irradiator. This is the highest flux point in the core. The radial flux at the fuel centerline is given in Table A.5.1.

Table A5.1

Flux Estimates

Location	**Flux at 90 kW (nt/cm^2·s)**	**Flux at 250 kW (nt/cm^2·s)**
A - Ring (Core Peak Flux)	$2.95 \cdot 10^{12}$	$8.18 \cdot 10^{12}$
B - Ring (Axial peak)	$2.66 \cdot 10^{12}$	$7.36 \cdot 10^{12}$
C - Ring (Axial peak)	$2.59 \cdot 10^{12}$	$7.20 \cdot 10^{12}$
D - Ring (Axial peak)	$2.63 \cdot 10^{12}$	$7.32 \cdot 10^{12}$
E - Ring (Axial peak)	$2.34 \cdot 10^{12}$	$6.52 \cdot 10^{12}$
F - Ring (Axial peak)	$1.83 \cdot 10^{12}$	$5.07 \cdot 10^{12}$
G - Ring (Axial peak)	$1.27 \cdot 10^{12}$	$3.55 \cdot 10^{12}$
Core Axial and Radial Average	$1.49 \cdot 10^{12}$	$4.14 \cdot 10^{12}$

A.6 Gamma Environment

Preliminary calculations of the gamma dose environment were made using linear extrapolation from our current gamma dose environment of 0.7 mR/hr at 100 kW, 2.5 times this flux results in 1.75 R/hr at 250 kW. However the gamma dose is not linear, due to build-up. An additional calculation was made base upon typical reactor data to support the estimation made above. Utilizing the four energy groups, the principle of superposition, and assuming a point source with build-up, a dose rate was calculated from the following relationships. The assumptions result in a very conservative estimation of exposure dose.

$$\phi_b = \frac{S}{4\pi R^2} B_p(\mu R) e^{-\mu R}$$

$$\dot{X} = 0.0659 \sum_i I_i E_i \left(\frac{\mu_a}{\rho}\right)^{\text{air}}$$

where

Fission rate (250 kW) = 7.81×10^{15} gamma rays/sec

B_p = Build-up factor

S = Source

R = 21 ft distance from source

μ = attenuation coefficient for water

$\dot{X}$ = dose rate (m/hr)

E = energy group (MeV)

ϕ = gamma flux (photons / $cm^2 \cdot s$)

$\mu a/\rho$ = mass attenuation coefficient for air (cm^2/g)

Substituting in the values for these constants we have:

For the dose at the surface of the reactor pool water and dose rate calculation:

Energy Group (MeV)	# of γ (Prompt and delayed γ per fission)	γ Flux ϕ 100 kW	γ Flux ϕ 250 kW	μ_a/ρ	$\Delta\gamma$ (100 kW) μR/hour	$\Delta\gamma$ (250 kW) μR/hour
0-1	8.4	$5.76 \cdot 10^{-9}$	$1.44 \cdot 10^{-8}$	0.00300	$1.0 \cdot 10^{-11}$	$2.7 \cdot 10^{-11}$
1-3	3.3	0.03	0.65	0.00256	$1.1 \cdot 10^{-3}$	$3.6 \cdot 10^{-3}$
3-5	0.40	18.76	46.92	0.00220	0.11	0.43
5-7	0.046	121.39	303.57	0.00206	0.89	3.91
Total					1.00	4.35

The gamma dose rate at the surface of the 250 kW reactor core is estimated as 4.35 mR/hr.

A.7 Reactivity Defect by Xenon Poisoning

Xenon poisoning in the core of the reactor is given by the equations below. First the equilibrium Xenon poisoning case will be calculated, then the time to equilibrium will be examined, and finally the poisoning at various times will be found.

- Xenon poisoning:

$$\Delta\rho \approx -\frac{X \cdot \sigma_X}{\Sigma_a} = -\frac{\sigma_X \cdot (\gamma_I + \gamma_X) \cdot \Sigma_f \cdot \phi}{(\lambda_X + \sigma_X \cdot \phi) \cdot \Sigma_a}$$

Time to Xenon equilibrium have the following equation set:

- Production of Iodine:

$$\frac{\partial I}{\partial t} = \gamma_I \cdot \Sigma_f \cdot \phi - (\sigma_I \cdot \phi + \lambda_I) \cdot I$$

Because σ_I is small we get:

$$\frac{\partial I}{\partial t} = \gamma_I \cdot \Sigma_f \cdot \phi - \lambda_I \cdot I$$

- Production of Xenon:

$$\frac{\partial X}{\partial t} = \gamma_X \cdot \Sigma_f \cdot \phi + \lambda_I \cdot I - \lambda_X \cdot X - \sigma_X \cdot \phi \cdot X$$

- Xenon poisoning verses time:

$$X = \frac{(\gamma_I + \gamma_X) \cdot \Sigma_f \cdot \phi}{\lambda_X + \sigma_X \cdot \phi} \cdot \left[1 - e^{-(\lambda_X + \sigma_X \cdot \phi) \cdot t}\right] - \frac{\gamma_I \cdot \Sigma_f \cdot \phi}{\lambda_X - \lambda_I + \sigma_X \cdot \phi} \cdot \left[e^{-\lambda_I \cdot t} - e^{-(\lambda_X + \sigma_X \cdot \phi) \cdot t}\right]$$

- At equilibrium we have:

$$X = \frac{(\gamma_I + \gamma_X) \cdot \Sigma_f}{\lambda_X / \phi + \sigma_X}$$

- Determining Σ_a:

$$\frac{\Sigma_f}{\Sigma_a} = \frac{\Sigma_f}{\Sigma_{aF}} \cdot \frac{\Sigma_{aF}}{\Sigma_a} = \frac{\eta}{\upsilon} \cdot f$$

So:

$$\Sigma_a = \frac{\Sigma_a}{\Sigma_f} \cdot \Sigma_f$$

Where we assume the following properties:

$\Delta\rho$ = change in reactivity (ρ\$)
X = xenon (atoms)
I = iodine (atoms)
Σ_f = macroscopic fission cross section (cm^{-1})
Σ_a = macroscopic absorption cross section (cm^{-1})
Σ_{aF} = macroscopic fuel absorption cross section (cm^{-1})
η = neutrons liberated per neutrons absorbed (2.073)
υ = average number of neutrons per fission (2.42)
f = thermal utilization (0.7241)
σ_X = microscopic cross section xenon ($2.65 \cdot 10^{-18}$ cm^2)
σ_I = microscopic cross section iodine (cm^2)
γ_X = fission yield of xenon (0.00237)
γ_I = fission yield of iodine (0.0639)
λ_X = half life xenon ($2.1 \cdot 10^{-5}$ s^{-1})
λ_I = half life iodine ($2.87 \cdot 10^{-5}$ s^{-1})
ϕ = neutron flux ($4.1 \cdot 10^{12}$ $nt/cm^2 \cdot s$)
t = time (s)

Xenon production will reach 95% of equilibrium in 34 hours operating at 250 KW. The equilibrium reactivity defect is -\$2.01. The flux used in these calculations is core average flux from A.5.

A.8 Reactivity Defect by Negative Temperature Coefficient

The temperature coefficient of the University of Utah TRIGA nuclear reactor is -$9.5 \cdot 10^{-5}$ $\Delta k/k \cdot {}^{\circ}C$ or -0.0136 $\rho\$/{}^{\circ}C$. In order to calculate the reactivity defect the average fuel temperature of the reactor must be known. Appendix A12 contains the thermo hydraulic conditions in the core. This maximum temperature expected in the peak fuel element is 440 °C and the average expected temperature is 222 °C , so the reactivity defect is now given by:

$$\Delta\rho = \alpha \cdot \Delta T$$

Where we assume the following properties:

$\Delta\rho$ = change in reactivity ($\rho\$$)
α = prompt temperature coefficient at 50 °C (-0.0136 $\rho\$/{}^{\circ}C$)
ΔT = change in temperature (220 - 20 °C) 202 °C

So as result $\Delta\rho$ is -\$2.75 of reactivity defect for 250 kW of power at steady state operation.

A.9 Reactivity Introduced by Cooling Water

The reactivity introduced by cooling water is analyzed two different ways. First the fuel is assumed to instantaneously drop 20 °C as result of contact with cold coolant water. The second calculation uses a lumped capacitance transient analysis. The lumped capacitance analysis assumed that the ratio of conductive to convective heat transfer of the fuel is very large. This is very conservative and predicts much faster cooling of the fuel than will actually occur. The lumped capacitance analysis shows that the time scale under which the cooling water adds reactivity over a very long period of time.

A sudden drop of 20 °C in the UUTR will cause a positive reactivity insertion due to the prompt negative temperature coefficient. The insertion of reactivity is given by the following relation.

$$\Delta\rho = \alpha \cdot \Delta T$$

Where we assume the following properties:

$\Delta\rho$ = change in reactivity ($\rho\$$)
α = prompt temperature coefficient at 50 °C (0.0136 $\rho\$/°C$)
ΔT = change in temperature (20 °C)

So as result $\Delta\rho$ is \$0.27 of reactivity would be added to the reactor.

Now the characteristic time scale can be found by a lumped capacitance analysis. Here the Biot number that gives the ratio of convective to conductive heat transfer is given by,

$$Bi = \frac{h \cdot L}{k} << 1$$

where:

h = coefficient of convective heat transfer (1171.6 $W/m^2 \cdot °C$)
L = characteristic length, volume over surface area (0.0091 m)
k = thermal conductivity of the fuel (10.7 $W/m \cdot °C$)

The resulting Biot number is 0.996, and this predicts the time required for the fuel to cool, and is thus conservative.

The lumped capacitance analysis is represented by the equation:

$$t = \frac{\rho \cdot V \cdot c}{h \cdot S} \cdot ln\left(\frac{\theta_1}{\theta}\right)$$

Here the constants are given as:

t = time (s)
ρ = density (5818.2 kg/m^3)
V = volume (0.0003965 m^3)
c = heat capacity (362 J/kg)
S = Surface area (0.04357 m^2)
θ_1 = difference of initial temperature and fluid temperature (90 °C)
θ = difference of fuel and fluid temperatures (85, 80, 75, and 70 °C)

The results of this calculation are listed in table A.9.1 below.

ΔT (°C)	**Time (s)**	**Δρ$ ($)**	**Insertion Rate**
5	0.94	0.068	0.073 $/s
10	1.93	0.136	0.071 $/s
15	2.98	0.204	0.068 $/s
20	4.11	0.272	0.066 $/s

Table A.9.1

In all cases the reactivity added is less than $0.10 per second.

A.10 Excess Reactivity

The excess reactivity of the UUTR core is determined primarily by xenon poisoning and the negative temperature coefficient of reactivity, and to a lesser extent by fuel burnup and experimental conditions. Xenon poisoning accounts for a reactivity defect of -$2.01 at equilibrium. The negative temperature coefficient of reactivity causes a reactivity defect of -$2.75 at maximum licensed power. Burnup and experimental requirements require approximately -$1.00 for the reactors operation over an extended period of time with out changing the fuel. Thus for 10 hours of operation an estimated $5.76 of cold clean excess reactivity is needed for operation of the UUTR.

A.11 Coolant Flow Rates

In this section the fuel element temperatures in the UUTR core at 250 kW are calculated from natural convection relations for a core average temperature, and for core maximum temperature (hot channel factor of 1.7). As detailed in the UUTR Safety Analysis Report Appendix II and III (1985) the reactor system can operate at steady state power levels up to 2000 kW before departure from nucleate boiling occurs for an 80 element core. Using an the same basis as in the 1985 SAR the hot spot in the fuel is 440 °C well below the 800 °C safety limit for stainless steel clad fuel. The aluminum-clad fuel will not be used in the 250 kW core configuration. The method used to determine the thermal characteristics of the core follows.

An 80 element hexagonal pitch core contains twice as many trifoliate coolant channels as there are elements. The hot channel and hot spot factors used for the calculation are:

Hot channel (peak to average) = 1.7

Axial hot spot (peak to average) = 1.5

An additional conservative assumption:

•7% uncertainty in the power measurements (268 kW, half of the 15 % allowable).

•core depth of 20 feet and the altitude of the University of Utah, the pressure at the core is 144.8 Pa (21 psia) and a water boiling temperature of 109.4°C (229 °F).

The wetted perimeter of the fuel is based on the triflute shaped channel (equilateral triangle with 3.937 cm pitch (1.55 in) and the radius of the fuel is 1.87 cm (0.735 in). The hydraulic diameter is

$$De = \frac{4\left[0.5 \cdot \frac{\sqrt{3}}{2}(1.55)^2 - .5\pi(0.735)^2\right]}{0.5 \cdot 2\pi \cdot 0.735} = 0.338in(0.858cm)$$

The pressure drop through the channels is interpolated between the conditions for a set of parallel plates and that for a pipe, cross flow between the channels is ignored.

$$\Delta p = 4f\left(\frac{L}{De}\right)\left(\frac{\rho v^2}{2}\right)$$

where f is the Fanning friction factor. In laminar flow,

4f = 96/Re for plates and 4f = 64/Re for a pipe.

For these channels 4f=80/Re (interpolated between plate and pipe).

The general equations used are:

$$q = \frac{1}{4f}\frac{\Delta\rho \cdot Af \cdot \Delta T \cdot g \cdot De \cdot Cp \cdot \Delta T}{v}$$

$$v = \frac{\Delta\rho \cdot g \cdot De^2}{80 \cdot \mu_{ave}}$$

where

q = heat produced by the reactor (250 kW)
v = velocity
ρ = fluid density
μ = fluid viscosity
μ average fluid viscosity in the channel
Af = flow area 1.23 cm^2 (0.19 in^2) per channel
De = effective diameter 0.858 cm (0.338 in)
Cp = specific heat 4.18 J/Kg K @ 20 °C,
ΔT= temperature difference between top and bottom of the channel
Δρ = density difference between top and bottom of the channel
g = acceleration due to gravity

The following properties of water were used.

Table A.11
Water Properties

T (K)	Density (kg/m^3)	Specific heat (J/kg K)	Viscosity (Pa s)
293	1000	4180	101 x 10^{-5}
333	985	4180	47.1 x 10^{-5}
373	961	4220	28.2 x 10^{-5}

The above relationships result in a volumetric flow rate of 0.0016 m^3/sec, and a mass flow rate of 1.6 kg/sec and a coolant velocity of 0.18 m/s

A.12 Fuel Temperatures

Using the coolant flow rate and velocity, the temperatures for fuel cladding, and centerline were estimated using the following relationships

$$T_c = \frac{q}{2 \cdot \pi \cdot k_c \cdot L} ln\left(\frac{R}{R_{in}}\right) + T_s$$

$$T_{fs} = \frac{q''_S}{h_{gap}} + T_c$$

$$T_{fc} = \frac{q''_V \cdot R_{in}^2}{4 \cdot k_f} + T_{fs}$$

$$T_{avg} = \frac{2}{\pi}\left(\frac{q''_V \cdot R_{in}^2}{8 \cdot k_f} + T_{fs}\right)$$

Using the following properties and results:

T_c = Interior clad temperature
T_{fs} = Fuel surface temperature
T_{cl} = centerline temperature
T_{ave} = average clad temperature
h_{gap} = heat transfer coefficient (6000 $W/m^2 \cdot K$)
R = cladding radius (0.0187 m)
R_{in} = fuel radius (0.01815 m)
L = heated channel length (0.381 m)
k_c = cladding coefficient of conductivity (13.4 $W/m \cdot K$)
k_f = fuel coefficient of conductivity (18.52 $W/m \cdot K$)

The results are:

Table A.12

Fuel and Water temperatures

Parameter	250 kW	100 kW
Water temperature °C	40	34.4
T_S, clad surface temperature	198.89	147.22
T_C, interior clad temperature	200	147.78
T_{fs}, fuel surface temperature	347.22	208.89
T_{fc}, max. fuel temperature	440 °C	246.11
Exit water temp	58.33	47.22
Peak heat flux (kW/m^2)	18.3	7.26

So the maximum centerline fuel temperature in the hottest fuel element is 440 °**C,** and the core has an average temperature of 220 °C. This result provides a safety factor of [1000 °C. / 440 °C. =]2.3 before onset of possible fuel damage

The assumptions made in these calculations are as follows:

(1) Physical properties are averaged for entire temperature range

(2) Physical properties are isotropic

(3) Bulk water temperature is averaged and held constant

(4) Cladding thickness and fuel size are uniform

(5) Reactor power is distributed uniformly throughout fuel

(6) peak power factor of 1.70

(7) Infinite cylinder

(8) k_c is independent of temperature

A.13 Decay Heat Calculations

The current UUTR system for 250 kW does not have a secondary cooling system. Thus, the primary tank water must absorb the entire 250 kW heat flux. An administrative limit to the tank bulk water temperature has been set at 40 °C. This limit was chosen because the deionization resins begin to degrade above 60 °C, and immersion in temperatures above 40 °C (104 °F) can cause scalding. The following analysis calculates the length of time the reactor can be operated before exceeding the 40 °C temperature limit assuming that the reactor coolant starting temperature of 20 °C. This time period is calculated from the following equation.

$$t = \frac{Cp \cdot \Delta T \cdot m}{Q}$$

Where the constants are defined as:

t = time (s)
Cp = specific heat of water (4180 J/kg·K)
ΔT = temperature increase (20 °C and 1 °C)
m = water mass (33,270 kg)
Q = reactor power (250 kW)

Thus the time required to reach 40 °C from 20 °C is 3 hours assuming an average water temperature increase of 1 °C FOR every 9 minutes of operation at 250 KW.

After shutdown the reactor power level immediately drops to 48 kW and then to 1 watt in approximately 15 minutes. Forced cooling is not required after shutdown. Without any auxiliary cooling the bulk water temperature will return to 20 °C in about 10 days via passive heat loss to the reactor room environs.

APPENDIX B- TABLE OF CONTENTS

B.1 Production of Ar-41 in the Experimental Facilities

Argon-41 production and release from the reactor's dry tube systems was determined as follows.

The governing equation is:

$$\frac{dN}{dt} = \Sigma \cdot \phi - \lambda \cdot N$$

$$N_{41} = \frac{\phi N_{40} \sigma_{40}}{\lambda_{41}} \left(1 - e^{-\lambda_{41} t}\right)$$

Where we assume the following properties:

N = atom density of argon in air
V_1 = reactor room volume = $5.65 \cdot 10^8$ cm^3
V_2 = volume in dry tube (diameter 3.8 cm and length 36 cm) = 1633 cm^3
N_A = Avogadro's number ($6.02 \cdot 10^{23}$ atoms/mol)
ϕ = average neutron flux ($4.136 \cdot 10^{12}$ nt/$cm^2 \cdot$s)
u = air exhaust flow velocity (50.8 cm/s)
S = exhaust flow cross sectional area (7432.2 cm^2)
t = time (s)

For Argon:
0.94 mole % in air
0.99 Ar 40 natural abundance
Density ($3.85 \cdot 10^{-7}$mol Ar /cm^3)
A = atomic mass (39.948 g/mol)
σ = microscopic cross section ($6.5 \cdot 10^{-25}$ cm^2)
λ = decay constant ($1.058 \cdot 10^{-4}$ sec) half life of Ar-41 **at** 1.82 hours
Σ = macroscopic cross section ($115.66 \cdot 10^{-9}$ cm)

At 1.0 hours the Ar-41 concentration is 3.026 x μCi/cm^3 for the volume of air in the experimental facility. The saturation activity for Ar-41 is 7.16 μCi/cm^3 for the same volume. The diffusion coefficient is ~ 1x 10^{-4} m^2/sec. The air in the dry tube is essentially stagnant, and Ar-41 behaves thermodynamically as Ar-40 so there is no driving force except temperature gradients. Assume that the of the mole fraction of Argon in the reactor room is zero, and using a diffusion coefficient of 0.0136 cm2/sec, for perfect mixing then the Ar-41 concentration is 4.67×10^{-6} Ci/cm^3. coefficient of 0.0136 cm2/sec, assuming perfect mixing the Ar-41 concentration is 4.67×10^{-6} Ci/cm^3. However to achieve perfect mixing takes 800 hours, and Ar-41 has a half life of 1.82 hours, Therefore negligible amounts of Ar-41 is released from the dry tube.

The pneumatic transfer system has twice the length of tubing before it releases into the fume. Based on the same assumptions used for the dry tube, negligible amounts of Ar-41 are released from the pneumatic transfer system. By 10CFR20 the derived air concentration (DAC) of Ar-41 for a restricted area is $3.0 \cdot 10^{-6}$ μCi/ cm^3 ($1.0 \cdot 10^{-8}$ μCi/ cm^3 unrestricted). There are minimal effects from increasing the power on the Ar-41 released for the experimental facilities.

Seader JD, Henly E, Separation Process Principles John Wiley and Sons, New York USA 1998

B.2 Ar-41 Production in Pool Water and Release

Using the approach detailed in UUTR SAR 1985, the Ar-41 concentration in the reactor room air from the activation of Argon - 40 dissolved in the water for 250 kW operation is

$(1.11 \times 10^{-8}\,\mu Ci/cm^3)$ at 250 kW

And Ar-41 released through the ventilation system is

$9.25 \times 10^{-3}\,\mu Ci/sec$ at 250 kW.

Estimated dose rate, based on conversion factors for submersion in a semi-infinite cloud of radioactive noble gases is

2.44×10^{-12} Sv per hour $(2.44 \times 10^{-14}$ REM/hour) (ICRP publication 30).

So the coupled equations are of the form:

$$\frac{dN_{41}|_1}{dt}\cdot V_1 = V_1 \cdot N_{40}|_1 \cdot \sigma_{40} \cdot \phi + u \cdot N_{41}|_2 - u \cdot N_{41}|_1 - \lambda \cdot N_{41}|_1 \cdot V_1 - V_1 \cdot N_{41}|_1 \cdot \sigma_{41} \cdot \phi$$

$$\frac{dN_{41}|_2}{dt}\cdot V_2 = u \cdot N_{41}|_1 - u \cdot N_{41}|_2 + f_{32} \cdot N_{41}|_3 \cdot V_3 - f_{23} \cdot N_{41}|_2 \cdot V_2 - \lambda \cdot N_{41}|_2 \cdot V_2$$

$$\frac{dN_{41}|_3}{dt}\cdot V_3 = f_{23} \cdot N_{41}|_2 \cdot V_2 - f_{32} \cdot N_{41}|_3 \cdot V_3 - q \cdot N_{41}|_3 - \lambda \cdot N_{41}|_3 \cdot V_3$$

The steady state equations are:

$$0 = V_1 \cdot N_{40}|_1 \cdot \sigma_{40} \cdot \phi + u \cdot N_{41}|_2 - u \cdot N_{41}|_1 - \lambda \cdot N_{41}|_1 \cdot V_1 - V_1 \cdot N_{41}|_1 \cdot \sigma_{41} \cdot \phi$$

$$0 = u \cdot N_{41}|_1 - u \cdot N_{41}|_2 + f_{32} \cdot N_{41}|_3 \cdot V_3 - f_{23} \cdot N_{41}|_2 \cdot V_2 - \lambda \cdot N_{41}|_2 \cdot V_2$$

$$0 = f_{23} \cdot N_{41}|_2 \cdot V_2 - f_{32} \cdot N_{41}|_3 \cdot V_3 - q \cdot N_{41}|_3 - \lambda \cdot N_{41}|_3 \cdot V_3$$

The volume flow rate of water through the core of the reactor was found by the following heat balance equation (note that the hot channel factor is absent in this equation):

$$u = \frac{Q}{\rho \cdot C_p \cdot \Delta T}$$

The saturation concentration of Ar-40 in the water of the reactor is determined by Henry's Law:

$$X = \frac{P}{K} \quad \text{And,} \quad N_{40}|_1 = \frac{\rho \cdot N_A \cdot X}{A}$$

The fractions of Ar-41 that enter and leave solution in the tank water are found by the following set of equations:

The movement of a diffusing particle can be described by the equation:

$$|\Delta x| = \sqrt{2 \cdot D \cdot t}$$

Less than half the Argon-41 atoms at the coolant water surface will displace upward and leave the tank water . Because of this, an upper limit on the 'source depth' can be set at 2.75 x 10^{-3} cm. Thus, the limit of Argon that can reach the surface is given by:

$$f_{23} = \frac{|\Delta x|}{2 \cdot h}$$

Now we can relate the fraction of Argon leaving the tank water with the fraction entering the tank water by the following relation:

$$f_{23} \cdot N_{40}\big|_2 \cdot V_2 = f_{32} \cdot N_{40}\big|_3 \cdot V_3$$

Rearranging we get:

$$f_{32} = \frac{f_{23} \cdot N_{40}\big|_2 \cdot V_2}{N_{40}\big|_3 \cdot V_3}$$

This model and the following assumptions were used to determine the Ar-41 concentration in the reactor room air. The variables and constants used in these equations are:

$N_{41}\big|_1$ = Ar^{41}_{18} Atom Density In Core (atoms / cc)
$N_{41}\big|_2$ = Ar^{41}_{18} Atom Density In Tank (atoms / cc)
$N_{41}\big|_3$ = Ar^{41}_{18} Atom Density In Reactor Room (atoms / cc)
$N_{40}\big|_1$ = argon 40 in tank water $\left(8.05 \text{ x } 10^{15} \text{ atoms / cc}\right)$
V_1 = core water volume (12400 cc)
V_2 = reactor tank water volume ($3.57 \cdot 10^7$ cc)
V_3 = reactor room volume ($5.65 \cdot 10^8$ cc)
σ_{40} = microscopic cross section Ar-40 ($6.5 \cdot 10^{-25}$ cm^2)
σ_{41} = microscopic cross section Ar-41 ($5.0 \cdot 10^{-25}$ cm^2)
ϕ = core average neutron flux ($4.136 \text{x} 10^{12}$ nt/$cm^2 \cdot$s)
q = exhaust system volume flow (653650 cc/s)
u = cooling water through core volume flow (3350 cc/s)
f_{23} = exchange fraction from tank water to reactor room ($3.6 \cdot 10^{-6}$ 1/s)

f_{32} = exchange fraction from reactor room to tank water ($7.27 \cdot 10^{-11}$ 1/s)
λ = decay constant Ar-41 (0.0001058 1/s)
Q = reactor power (250 kW MW)
C_p = specific heat of water (4190 J/kg·K)
ρ = density of water (980.1 kg/m^3)
ΔT = temperature difference of cooling water through core (40 °C)
X = mol fraction of Argon in water ($2.38 \cdot 10^{-7}$)
P = partial pressure of argon above water (6.9 mm of Hg)
K= Henry's constant for argon ($2.9 \cdot 10^{7}$)
ρ = density of Ar (1.7837 kg/m^3)
N_A = Avogadro's number ($6.022 \cdot 10^{23}$ atoms/gram)
A = atomic mass of Ar (39.948 gram/mol)
h = tank height (761.0 cm)
Δx = displacement ($2.75 \cdot 10^{-3}$ cm)
D = diffusion coefficient ($1.5 \cdot 10^{-5}$ cm^2/s)
t = time (1.0 s)

The physical assumptions made in these equations are:

(1) Complete mixing.
(2) Properties averaged over position.
(3) Ar-40 level is constant in the tank water.
(4) Transient conditions are ignored because steady state solution is a maximum.

B.3 Argon-41 Plume Calculations

The maximum release from experimental facilities of Ar-41 in the reactor room exhaust air was calculated to be at a concentration of ~ 0 μCi/ml. In addition, from B.2, a steady concentration from the pool water of 1.11×10^{-8} μCi/cm³ was predicted. Thus the maximum total is 1.11×10^{-8} μCi/cm³.

A typical method for estimating the hazard to the community of a constant source of gaseous activity is to assume a Gaussian form for the plume dispersion, pertinent values for the diffusion parameters and a mean wind velocity corresponding to UUTR location meteorological conditions. This method yields a conservative account since it is improbable that such stable parameters will remain constant over long periods of time. Any instability in the meteorological conditions will enhance mixing and dispersal and reduces the dose received at any given location away from the source.

Detailed site-specific estimates of the release of radioactivity can only be made from measurements at the site. In the present case, the very low level of the hazard does not justify such a procedure. However, a conservative calculation demonstrates that hypothetical dose rates are below those derived air concentrations and effluent release values.

It should be noted that the rate of Ar-41 released, quoted previously and used in the calculations, is estimated for an experimental condition that is unlikely to arise frequently. Typically, the release rate will be lower.

The generalized Gaussian Plume Model has the form,

$$X = \frac{2 \cdot Q}{\pi \cdot u \cdot \sigma_y \cdot \sigma_z} \cdot exp\left[\frac{-1}{2}\left(\frac{y^2}{\sigma_y^2} + \frac{h^2}{\sigma_z^2}\right)\right]$$

and,

$$\sigma_y = \frac{C_y \cdot x^{1-n/2}}{2}, \quad \sigma_z = \frac{C_z \cdot x^{1-n/2}}{2}$$

where,

X = concentration of Ar-41 (μCi/m³)
Q = source strength (0.025 μCi/s)
u = mean wind speed (1.0 m/s)

x = windward distance from source (350 m)
y = crosswind distance from the plume axis (0 m)
h = source height (22 m)
σ_y, σ_z = cloud centerline concentration standard deviation in y, z (32.4, and 5.66)
C_y, C_z = virtual diffusion coefficients (0.40, and 0.07 $m^{1/4}$)
n = stability parameter (0.5)

The prevailing daytime wind direction in this region of the Salt Lake Valley is from the west to the east, traveling up Red Butte Canyon so that the cloud will be transported east up the Wasatch mountain range away from the center of the Campus. For poor dispersion conditions, assumed here, the maximum dose is conservatively assumed to be received at a location close to the source and at a similar height. The nearest building of similar height is the Mines Building, a six story structure which houses the College of Mining located about 350 meters south-west of the reactor site (this direction is essentially into the prevailing day time winds, but downwind for some prevailing night time winds). The Ar-41 concentration for this position assuming it coincides with the cloud centerline (i.e. assume conservatively that $y = h = 0$) and $x = 350$ meters is found to be $1.9 \cdot 10^{-9}$ μCi/ml, which is five times less than the unrestricted limits of $1.00 \cdot 10^{-8}$ μCi/ml. This conservative estimate does not account for "building dilution," dilution by other exhaust fans operating on the Merrill Engineering roof (estimated to provide further dilution by at least a factor of 10 at all times), radioactive decay of the Argon-41 during transport, or the fact that the reactor will be operated approximately 20 hours per month.

B.4 Nitrogen Production and Transport in the Pool Water

Using the approach detailed in UUTR SAR 1985, the N-16 concentration in the reactor room air from the O-16(n,p)N-16 is found to be ($1.75 \times 10^{-9}\ \mu Ci/cm^3$) at 250 kW. These calculations began with an estimation of the N-16 production from the core as the result of fast neutron reactions, O-16(n,p)N-16. Gamma flux from the core region will be accounted for in the calculations of the distributed source. Nonetheless, the core source term is fundamental to the development of the distributed source calculations. Steady state conditions were assumed. From the gamma flux at the surface, an additional dose will be calculated for a point two meters above the core. This point was chosen as the closest credible location of a worker for any substantial length of time during reactor operations.

The concentration of N-16 per cm^3 of water as it leaves the core is given by the following equation

$$N_N = \frac{\phi N_o \sigma_o}{\lambda_N}\left[1 - e^{-\lambda_N t}\right]$$

N_N = atom density of N-16 (atoms/cm^3)
σ_O = microscopic cross section for $O_{16}(n,p)N_{16}$ reaction ($5.5 \cdot 10^{-26}$ cm^2)
N_O = atom density of O-16 in core coolant water ($3.35 \cdot 10^{22}$ atoms/cc)
λ_N = decay constant of N-16 (0.09712 s^{-1})
ϕ = neutron flux at 250 kW (E > 10MEV) (nt/ $cm^2 \cdot s$)
t = average time of exposure in reactor .

The average exposure time in the reactor core is given by

$$t = \frac{V_c}{v_1}$$

$V_c = 1.24 \times 10^4$ cm^3
v_1 = 1522 cm^3/sec ($v = Q/C_p \Delta T \rho$)

Then t =8.15 sec, and $N_N = 9.4 \times 10^6$ atoms N-16/cm^3. The average vertical velocity of the water rising from the 250 kW core is ~28.8 cm/sec. The transport time for the water to move 21 feet (640 cm) to the surface is 22 seconds and $N_{N\text{-}sur}$ = 1831 atoms N-16/sec reach the surface.

B.5 Nitrogen Diffusion into the Air

To estimate the N-16 escaping the pool water into the reactor room air, we use the equations developed in sections B.1, B.2, and B.4.

$$0 = f_{23} \cdot N_{N_{16}/2} \cdot V_2 - \lambda \cdot N_{N_{16}/3} \cdot V_3$$

and:

$$N_{N_{16}/3} = \frac{f_{23} \cdot N_{N_{16}/2} \cdot V_2}{\lambda \cdot V_3} \quad ;$$

where:

$$f_{23} = \frac{|\Delta x|}{2 \cdot h} = \frac{\sqrt{2 \cdot D \cdot t}}{2 \cdot h} \quad .$$

and where:

f_{23} = fraction of N-16 leaving the tank water (s^{-1}) (5.5×10^{-5})
$N_{N16/2}$ =3.2×10^{-4} density of N-16 in the upper 30 cm of the tank
$N_{N_{16}/3}$ = density of N_{16} in reactor room (atoms / cc)
V_2 = volume of N-16 distribution in tank (5603779 cm^3)
V_3 = volume of N-16 distribution in reactor room (141,250,000 cc)
λ = N-16 decay constant (0.09712 s^{-1})
D = diffusion coefficient ($4.5 \cdot 10^{-6}$ cm^2/s)
t = time (1.0 s)
h = depth of N-16 (30 cm) .

The amount, then, of N-16 leaving the tank water was calculated to be 7.6×10^{-9} atoms per sec per cm ^{3} in the air. The dose from these concentrations is found from the following relationship:

$$D = \frac{S_{16} \cdot \left(1 - e^{-\mu \cdot R}\right)}{2 \cdot \mu \cdot c} \quad .$$

Here the constants are defined as:

D = dose (mRem/hr)
S = activity of N-16
μ = linear absorption factor ($3.03 \cdot 10^{-5}$ /cm)
R = volume radius (300 cm)

c = dose conversion factor (160 $Bq \cdot cm^2/mRem/hr$)

The resulting dose rate is $7.1 \cdot 10^{-9}$ mRem/hr

B.6 Nitrogen-16 Dose From Cooling System

The cooling system is not employed during 250 kW because of the limited cooling capacity.

B.7 Beam Port Neutron and Gamma Flux

The beam ports will not be opened. Based on the present operating configuration, the neutron and gamma fluxes are negligible and the dose is zero. If operation of these beam ports is proposed, radiation assessments will be preformed and reviewed by the reactor safety committee and NRC if necessary.

B.8 Gamma Dose From In Core Experimental Facilities

Using the prompt and delayed gamma per fission and conservatively assuming : a point source at the end of the dry tube, attenuation, buildup for air is negligible and there is no curvature in the tubes : then Dose rate at the upper end of the dry tube and pneumatic tube is given in Table B8

Table B8

Gamma doses from the experimental facilities

Energy Group (MeV)	# of γ (Prompt and delayed per fission)	γ Flux ϕ Dry tube	γ Flux ϕ PTS	μ_a/ρ	$\Delta\gamma$ Dry tube	$\Delta\gamma$ PTS
0-1	8.4	2.66E-08	2.66E-08	0.0636	$1.0 \cdot 10^{-11}$	$2.7 \cdot 10^{-11}$
1-3	3.3	0.59	0.59	0.0357	$1.1 \cdot 10^{-3}$	$3.6 \cdot 10^{-3}$
3-5	0.40	14.67	14.67	0.0274	0.11	0.43
5-7	0.046	53.51	53.51	0.022	0.89	3.91
Total					0.485	0.485

At 250 kW the combined doses from the experimental facilities is conservatively estimated as 0.97 mR/hour.

B.9 Fission Product Inventory

An estimate of the UUTR's fuel inventory is made from the computer program RSAC-5. This inventory is based upon operating the reactor for 60 hours every six months for 5 years. This fission product and activated isotope inventory is also subdivided into groups.

Inventory:

Radiological Safety Analysis Computer Program (RSAC-5)
(RSAC-5E, Rev 5.1, 08/09/93) Date 12/05/97 Time 11:07
250 kW For 10 Hours Per Month For 2 Years and 1 Week Decay.

[redacted]

[redacted]

[redacted]

[redacted]

B.10 Gamma Dose From Ion Exchange Canister

To estimate the dose from ion exchange canisters at 250 kW, the actual measured dose is extrapolated from 90 kW to 250 kW. With the present dose rate measured from ion exchange resins at 0.15 mRem/hr taken directly after a one hour 90 kW run, the extrapolated dose for 250 kW is 0.416 mRem/hr. The ion exchanger for the UUTR are within the reactor room and are controlled and labeled as radiation sources for worker notification as necessary.

APPENDIX C- TABLE OF CONTENTS

C.1 Reactivity Accident

In this section two types of accidents are evaluated. These are a prompt excursion and a ramp insertion of reactivity. The scenarios under which these accidents occur are discussed in chapter 13.

C.1.1 Prompt Excursion

Several calculations were performed in order to determine the effect that a sudden large insertion of reactivity has on the University of Utah's TRIGA Reactor (UUTR.) The rapid insertion of reactivity is performed for four cases. These are:

Insertion of excess reactivity
Removal of highest worth control rod
Insertion of fuel into highest worth position
Removal of a maximum negative worth experiment

In each of these cases the maximum reactivity insertion possible will be determined and then used in the Fuchs-Nordheim model for the prompt excursion. From this model the maximum fuel temperature is determined with the corresponding reactivity insertion.

The Fuchs-Nordheim model is developed as follows. A coupled set of differential equations that relate power and temperature to time are given as:

$$\frac{dP}{dt} = \frac{(\rho - \alpha \cdot T) \cdot P}{L} \qquad \text{and} \qquad \frac{dT}{dt} = \frac{(P - P_0)}{C}.$$

Combining these equations we get:

$$\frac{dP}{dT} = \frac{(\rho - \alpha \cdot T) \cdot (C_0 + \gamma \cdot T) \cdot P}{L \cdot (P - P_0)}.$$

Solving this equation the following relationship is obtained:

$$L \cdot \left[(P - P_0) - P_0 \cdot ln\left(\frac{P}{P_0}\right)\right] = T \cdot \left[\rho \cdot C_0 + \frac{(\gamma \cdot \rho - \alpha \cdot C_0) \cdot T}{2} - \frac{\alpha \cdot \gamma \cdot T^2}{3}\right].$$

Maximum average fuel temperatures occur when:

$$\frac{dT}{dt} = 0 = (P - P_0)$$

So P=P_0 and Tmax are found from the expression:

$$T_{max} = \frac{3 \cdot \rho \cdot \left(\frac{\alpha \cdot C}{\gamma \cdot \rho} - 1\right)}{4 \cdot \alpha} \left\{ \left[1 + \frac{16 \cdot \frac{\alpha \cdot C}{\gamma \cdot \rho}}{3 \cdot \left(\frac{\alpha \cdot C}{\gamma \cdot \rho} - 1\right)^2} \right]^{1/2} - 1 \right\}$$

For the preceding formulas, we assumed the following properties:

ρ = reactivity above prompt critical
α = negative temperature coefficient ($1.77 \cdot 10^{-4}$ 1/°C
L = prompt neutron life time ($3.90 \cdot 10^{-5}$ s^{-1})
P_0 = reactor power (W)
T_{max} = maximum core temperature volume averaged (°C)
T = average core temperature (°C)
C = specific heat fuel ($1.27 \cdot 10^5 + \gamma \cdot (T_{max}-25)$ J/°C)
γ = change in specific heat per degree Celsius (143.0 J/°C^2)
t = time (s)

The amount of excess reactivity available for each of four scenarios above is determined as follows. In case 1 the insertion of the full excess reactivity allowed by technical specifications of $5.50 is instantly inserted into the core. In case II the highest worth control rod is $2.80 is instantly inserted into the core. For case III the highest worth fuel location of $2.25 (A ring) is instantly added to the core. In case IV an experiment with a negative reactivity of $3.00 is removed from a critical core.

By varying the reactivity insertion in the above equations, we can determine the associated reactor period, maximum reactor power output, and maximum fuel temperature. The values calculated for these reactor periods, maximum reactor powers, and maximum fuel temperature are given in Table C.1.1. Phase changes for stainless steel occurs at 530 °C and cladding failure occurs at 1150 °C.

Table C.1.1

Reactivity Insertion ($\rho_\$$)	Temperature (0 W)*	Temperature (250 kW)**
1.5	178	598
2.0	333	753
2.5	487	907
3.0	639	1059
3.5	790	1210
4.0	940	1360
4.5	1089	1509

Note: * FOR A reactor power = 0 W, fuel temperature = 20 °C.

** FOR A reactor power = 250 kW, xenon-free, and at a maximum fuel temperature 440 °C.

C.1.2 Ramp Insertion of Reactivity

A ramp withdrawal of a control rods is detailed in these calculations. Here it is assumed that all of the control rods are driven out of the core at the maximum rate. The equations used to describe this occurrence are:

$$\frac{dP(t)}{dt} = \frac{\dot{\rho} \cdot t - \beta}{\Lambda} \cdot P(t) + \lambda \cdot C(t) ,$$

$$\frac{dC(t)}{dt} = \frac{\beta}{\lambda \cdot \Lambda} \cdot P(t) - \lambda \cdot C(t) \text{, and:}$$

$$\frac{dT(t)}{dt} = \frac{P(t) - P_0}{Cp} .$$

Initially the reactor is assumed to be critical at 1.0 W, so the initial precursor population is found from:

$$C(0) = \frac{\beta \cdot P(0)}{\lambda \cdot \Lambda} , \quad \text{and} \quad T(0) = 20\ ^\circ\text{C}.$$

Here the constants are defined as;

$\dot{\rho}$ = rate of reactivity addition (0.00191 $\Delta k/k \cdot s$)

L = prompt neutron life time ($3.90 \cdot 10^{-5}$ s^{-1})
P(t)= reactor power at time t (W)
C(t) = precursor power (W)
Cp = specific heat fuel ($1.27 \cdot 10^5 + \gamma \cdot (T-25)$ J/°C)
γ = change in specific heat per degree Celsius (143.0 J/°C^2)
β = precursor percentage (0.007)
λ = mean precursor lifetime (0.405 1/s)
t = time (s)

The above equations were solved using a fourth order Runga-Kutta numerical code. The initial value was a power of 1.0 W, and the precursor number found from equilibrium conditions. The rate of reactivity insertion was found by dividing the core's excess reactivity by the control rod travel distance (38.1 cm) and then multiplying by the rod drive speed (1.016 cm/s). This resulted in exceeding 110% of the reactor's maximum licensed power in 4.12 seconds when the reactor would scram with a 0.25 second delay and the control rods would drop. The rod drop time was estimated at 0.5 seconds inserting $4.00 dollars of reactivity (shutdown margin plus safety rod). The temperature rise was found to be about 21.1 °C during this event due to the low integrated power. So the resulting maximum temperature of the fuel would be 41.1 °C.

C.2 Loss of Coolant Accident (LOCA)

C.2.1 LOCA by Cataclysmic Accident

The first accident scenario analyzed is the loss of coolant by the complete opening of both containment tank and the concrete pad to the soil. From Darcy's law we have:

$$q = \frac{\kappa}{L}\cdot(h_l - h_2), \quad t = \frac{h_l}{q} = \frac{L}{\kappa}, \text{ and } \quad t = \frac{h_l}{q} = \frac{L}{\kappa}$$

where we assume the following properties:

q = seepage velocity (m/s)
κ = Darcy's constant (0.005 cm/s for sand)
h_l = normal pool water height (6.4 m)
h_2 = empty pool water height (0 m)
L = minimum radius of hemisphere of saturated sand from full tank (4.15 m)
R = radius of tank (1.22 m)
ρ_r = percent porosity (20 % for sand)
t = time (s) ;

The time needed to drain all of the pool water is t = 23.1 hours, and the time needed to drain 6 inches below its normal level is 32.9 minutes.

C.2.2 LOCA by Loss of Coolant by Evaporation

The second accident scenario analyzed is the loss of coolant through evaporation of the tank water during operation of the reactor at 250 kW power. The analysis began with:

$$t = \frac{L_v}{P},$$

where we assume the following properties:

P = reactor power ($250 \cdot 10^3$ J/s)
L_v = Heat of vaporization of water ($2.26 \cdot 10^6$ J/kg)
t = time (s)

This equation predicts that it takes 9.04 seconds to evaporate away 1 kg of pool water. The mass of pool water is given by:

$$M = \rho \cdot V$$

and:

$$V = \pi \cdot R^2 \cdot h \ ,$$

where we assume the following properties:

M = mass of pool water (kg)
V = pool water volume (0.7117 m^3 for 6 inches -31.314 m^3 for whole tank)
R = tank radius (1.22 m)
h = Tank height (0.152 m for 6 inches -6.7 m for whole tank)
ρ = density (1000 kg/m^3) ;

As indicated by the calculations, a mass of water in the pool with 6 inches of height and 4 feet of radius is 711.68 kg. The time needed to boil away 711.68 kg of water is 107 minutes. The coolant loss rate is 0.11 kg/s. The total mass of the pool water is 31,314 kg, and the time needed to boil away all of the pool water is 78.6 hours. This calculation assumes the water is at the boiling point (~100 C) at the start and there is no heat loss to the tank environs, and that criticality can be sustained at this temperature.

C.2.3 Radiation Dose from Exposed Core

Examining both cases of pool water loss, evaporation and leakage, the Radiation Dose rate calculations are presented in Table C.2.3. The results in these tables assume that the reactor has been operating for 10 hours at a power level of 250 kW. Table C.2.3 shows the increasing exposure rates to the floor of the laboratory as the pool water drops at a rate of 0.4 meters per hour. In order to conservatively estimate these dose rates, the following calculations were performed. First, the decay power production is found by the expression:

$$P = P_0 \frac{\beta(1-\rho)}{\beta - \rho}$$

$$P = P_0 \cdot 5 \cdot 10^{-3} \cdot 27.43\left[\tau^{-0.2962} - \left(T_0 + \tau\right)^{-0.2962}\right].$$

$$\phi_b = \frac{S}{4\pi R^2} B_p(\mu R) e^{-\mu R}$$

$$\dot{X} = 0.0659 \sum_i I_i E_i \left(\frac{\mu_a}{\rho}\right)$$

The results of these calculations are displayed in the following table:

Table C.2.3 Dose from Decay Power

Time After Shutdown	**Water Height Over The Core**	**Buildup Due To Water**	***MAX Dose @ surface of the water mR/hour***
1 hour	605.8 cm	865.9	0.526
2 hours	571.5 cm	689.2	0.268
4 hours	503.0 cm	433.8	0.07
8 hours	366.0 cm	165.1	0.007
12 hours	229.0 cm	56.1	0.001
18 hours	23.6 cm	3.38	0.004
24 hours	0	1	2.48×10^{-7}
1 week	0	1	~
1 month	0	1	~

External dose from an exposed core can also be the result of activation of the structure and fission products. The majority of the UUTR structure is aluminum, which has such a short half-life that it would decay long before the core is exposed. However the core does contain a SS clad fuel and other fission products that emit gamma's that could result in an exposure. Assuming 1 mg of Co-60 is present as a contaminant in the core structure would result in an additional 8 R/hour at 24 cm for an exposed core.

C.2.4 LOCA Thermo-Hydraulic Calculation

In this section the fuel element temperatures in the UUTR core at 250 kW are calculated from natural convection relations after a loss of coolant accident. It is assumed that the reactor has been running at full power (250 kW) for 20 hours prior to the instantaneous and complete loss of reactor coolant. Natural convection calculations are made to determine the core average temperature, for core maximum temperature (hot channel factor of 1.7, fuel pin factor 1.33), the claddings average, and the claddings maximum temperatures.

In this case the increase in fuel temperature is calculated by assuming that all energy produced by decay power goes into heating the fuel's mass subtracting the heat transferred from the fuel surface. The change in fuel temperature only includes the convective heat transfer rate from the can be found from the shutdown power decay equation,

$$P = 0.65 \cdot P_o\left[(\tau - T_o)^{-0.2} - \tau^{-0.2}\right]$$

The fuel cladding temperature can be calculated from:

$$\Delta T = \frac{1}{Cp \cdot m}\left[\int_{T_0}^{t} P \cdot d\tau - A \cdot h \cdot T_s \cdot (t - T_0)\right]$$

In this case the only unknown variable is the time scale, t, below which decay heating dominates natural convection to the air, and the fuel temperature continues to rise. To determine this time scale an energy balance is used between decay heating and convection heat transfer is used.

$$A \cdot h \cdot (\Delta T + T_s) = P$$

In this energy balance there exists a time, t, where the convection cooling and decay heating exactly balance. This point in time represents the maximum temperature that the fuel will reach before convective cooling dominates and the fuel begins to cool.

In order to calculate coefficient of convective heat transfer we must examine the natural convection relations. The flow regime in this case, in contrast to previous calculations (in water), is laminar. In this case we first calculate the Grashoff number from the following:

$$Gr_L = \frac{g \cdot B \cdot (T_S - T_\infty) \cdot L^3}{\nu^2}$$

where:

$$B = \frac{-1}{\rho} \cdot \left(\frac{\rho_\infty - \rho}{T_\infty - T_s} \right).$$

The Nussult number is given by:

$$Nu_L = \frac{4}{3} \cdot \left(\frac{Gr_L}{4} \right)^{1/4} \cdot g(Pr)$$

where:

$$g(Pr) = \frac{0.75 \cdot Pr^{1/2}}{\left(0.609 + 1.221 \cdot Pr^{1/2} + 1.238 \cdot Pr\right)^{1/4}}$$

And the heat transfer coefficient by:

$$h = \frac{k \cdot Nu_L}{L}$$

Substituting in the appropriate constants we have:
(all thermo-hydraulic constants evaluated at $(T_S+T_\infty)/2 = 450$ K)

- P = decay power (W)
- P_0 = reactor power at shutdown (250 kW)
- τ = time from run start to measured heating (s)
- T_0 = time from run start to shutdown (72,000 s)
- t = time to convective domination (s)
- $Cp \cdot m$ = Specific heat and mass (157,600 W/K)
- ΔT = average change in fuel temperature (K)
- Gr_L = Grashoff number
- $g(Pr)$ = dimensionless temperature gradient
- Nu_L = Nusselt number
- h = convective heat transfer coefficient ($W/m^2 \cdot K$)
- g = acceleration of gravity (9.81 m/s^2)
- L = heated flow channel length (0.381 m)
- A = fuel surface area (3.805 m^2)
- ρ_∞ = density of air at STP (1.25 kg/m^3)
- ρ = density of heated air (0.7741 kg/m^3)
- ν = viscosity of heated air ($32.39 \cdot 10^{-6}$ m^2/s)
- k = thermal conductivity of heated air ($37.3 \cdot 10^{-3}$ $W/m \cdot K$)

Pr = Prandtl number of heated air (0.686)

The heat generation verses the heat transfer rate balance each other at 41.3 minutes after reactor shutdown. The average fuel temperature increase is 95 °C, bring the average fuel and cladding temperatures to 333, and 274 °C respectively. Using a radial hot spot factor of 1.7, a pin radial factor of 1.33, and an axial factor of $\pi/2$ the maximum clad and fuel temperatures are 891.6 °C and 462 °C respectively. Both of these temperatures are below the fuel temperature limit of 450 °C and a cladding temperature limit of 1150°C.

The **conservative** assumptions made in these calculations are as follows:

(1) Physical properties are averaged for entire temperature range
(2) Physical properties are isotropic
(3) ambient air temperature is averaged and held constant
(4) Cladding thickness and fuel size are uniform
(5) peak power factor of $1.70 \cdot \pi/2 \cdot 1.33$
(6) Fluid flow is laminar
(7) heat transfer is neglected during decay heating

C.3 Mishandling or Malfunction of Fuel

The maximum credible accident for TRIGA reactors involves failure of the cladding of a single fuel element after extended reactor operations, followed by instantaneous release of the fission products directly into the air of the reactor room. Conservative assumptions made are 50% of the halogens and 100% of the noble gases, and 1% of the solids are released into the reactor room.

The fission product inventory of interest for is:
(Assuming 1 year at 1.1 MW, 0 time for decay – the doses calculated by a factor of 4)

[redacted]

For this accident at the UUTR two situations arise. In the first situation the doses to the restricted area are determined, and second to unrestricted areas.

Dose to Restricted Area:

The dose to the restricted area was obtained in an identical manner as in section C.1 with a different fission product inventory as given above. The resultant effective dose, assuming submersion is 5.83mR/ breath. Therefore the maximum occupational dose is 1.2 R to the worker in the reactor room for 10 minutes without respiratory protection following the cladding failure of a single fuel element in water. If evacuation occurs within 2 minutes, (The area radiation alarms would provide notification of this radiation level and evacuation for the reactor room can be made rapidly), then the effective dose drops to 235 mR. All of these doses are within the NRC guidelines for occupational exposure as stated in 10CFR20.1203.

Dose to Unrestricted Area:

Assuming the worst-case scenario

- all the fission products are released instantaneously out the stack (2.86 Ci, 1360 rem effective dose)
- individual is located at the centerline of the plume.
- Wind speed is 10 m/s (SLC, Utah)

The dose to **an unrestricted** individual is 0.0624 mRmem / min. The **maximum** allowed dose (**100 mrem**) in the unrestricted **area is** reached **within** 26 hours. If the wind speed is reduced to 1m/s. the allowed dose in the unrestricted area will be reached in 2 hours.

De Nevers N. Air Pollution Control Engineering, (pg 123) Mc Graw Hill, New York, USA 1985

EPA publication number 520/1-88-020: Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion Federal Guidance Report No. 11.

Shapiro J. Radiation Protection A guide for scientists and Physicians Ed 3, Harvard University Press, Cambridge, Mass. USA, 1990.

C.4 Loss of External Power

The loss of off-site electrical power does not impact the radiation safety and alarm equipment in the reactor room since an emergency battery power backup unit is always available. Even the failure of the emergency backup power has no nuclear related consequences. Below are listed the results of primary and emergency power losses. The consequences of each of these events will be discussed following the listed results.

Primary power loss will result in the following events:

1) Loss of control rod magnet power.
2) Loss of secondary cooling system.
3) Loss of reactor power monitoring channels.
4) Loss of Continuous Air Monitors (CAM).
4) Loss of crane power.
5) Loss of overhead lighting.
6) Loss of ventilation system.

Loss of emergency backup power:

1) Loss of Area Radiation Monitors (ARM).

Loss of control rod magnet power will cause the control rods to drop as with any other SCRAM input. The control rod design is such that the reactor is shut down with the highest worth control rod stuck out.

Loss of the secondary cooling system will have no impact on the ability of the primary cooling system to cool the reactor.

Loss of power monitoring channel does not affect reactor safety due to the SCRAM initiated at the loss of power. Additionally, if even the power monitoring channels were to momentarily flicker, then this would cause a SCRAM in the linear and percent power channels.

The loss of continuous air monitors will not have any significant impact on the safety of the reactor. This systems primary purpose is to detect the onset of leaking fuel. Under previous accident scenarios the limited extent of danger posed by operating the reactor with damaged fuel show that the safety issues associated with this event are small.

The loss of power to the overhead crane locks the crane in position so it can not be moved up or down.

The facility is equipped with emergency lighting and each has its own internal battery. This system is checked on a monthly basis.

Loss of emergency area radiation monitors is offset by the use of portable dose rate instruments that are always available in the laboratory.

Appendix D - Table of Contents

D.1 Pool Water

Ultrasonic water level

The base sends out pulses of ultrasonic waves that strike the water surface and are reflected back to the meter. The unit measures the transit time of an ultrasonic wave to and from the distance to the water surface. The unit has been configured to have a range of 6 inches from its detector to 12 inches from the detector (refer to the detector manual in the equipment filing cabinet). This 6 inch measuring span is sent from the detector as an analog voltage (the output of an 8 bit DAC) with 0 V for 6 inches and 10 V for 12 inches. The output impedance of this voltage (10k ohms) is used in conjunction with R60 to divide the output voltage by 2 for input to the OMEGA controller display unit. By using appropriate offset and gain constants in the controller, the actual water level is displayed in inches. The detector unit is powered from the ±12 V battery backup voltage. The detector contains a relay that can be used for the low water alarm in place of the float and microswitch presently in use. The controller also contains an output relay that could be wired to turn on a low water light on the console. Since the output DAC has only 256 steps, the maximum resolution of the water level meter is 0.023 inches or roughly 1/2 millimeter.

D.2 Power Monitoring Channels

Linear Power Channel

The range switch on the rod drive panel selects a shunt resistor that is connected across the input to the linear channel amplifier. The current from the compensated chamber flows through this resistor and produces the 0.1 V full scale input voltage for the amplifier. Q201 and Q202 are FETs with less than 1 pA of reverse current. They serve to clamp the input in the event of a transient voltage and protect the amplifier input. In the event of a fault, R212 limits the current through the clamps to less than 1 mA. A power resistor is used because it may have to drop the entire 800V from the high voltage power supply (about 0.7 W). R203 and C201 form a 50 Hz low pass filter to remove unwanted high frequency components from the signal. The filter is placed after the clamping circuit to prevent a large transient in the output voltage caused by the input voltage spike that occurs when the operator changes range (for a brief moment there is effectively no shunt resistor). U201 is configured for a non-inverting adjustable gain of about 10. The exact gain is determined by the thermal power calibration procedure, and is set by adjusting R206. Diodes CR203-206 serve to protect the output of U203 from accidental overload in conjunction with fuse F201.

R211 is the shunt resistor that converts the ion chamber current to a voltage, while CR207 and 208 are PIN diodes which protect the input to U202. U202 is low offset voltage bipolar op amp that is configured for a non inverting gain of ten. C205 sets the 3 dB bandwidth to 50 Hz. VR201 provides a measure of protection against over voltages on the output of the amplifier.

Integrated Power

The 0 - 1 V output signal from the linear channel is divided down to 0 - 10 mV by R401 and R402, and filtered by C401. This voltage is the input to the precision voltage to frequency converter (see the National Semiconductor Linear Data book under LM331 applications) formed by U401 and U402. An LT1012 was chosen for U401 because both low offset voltage (3011V max.) and low bias current (150 pA max.) were required. The second deck on the range switch is configured exactly like the first deck with decade resistors from 100 Ω at 100 kW to 100 mΩ at

0.1 W. The resistor selected, along with C402, determines the fundamental time constant of the voltage to frequency converter. The effective gain of the V/F can be adjusted using R406, the pulse current setting resistor. VR401 provides a ±5 V supply from the ±15 V supply for all of the following circuitry. The output pulse-frequency is divided by 1000 by U403, U404, and U405. One half of the dual one-shot U406 is used to turn each pulse into a 1 mSec downward going pulse. The LCD counters count each pulse as one Watt-Hour. This output signal is also rectified by a Shottky diode, filtered, and used as the ±5 V supply for the counters. The counters receive their power and signal through the same coaxial cable, eliminating extra wiring. U409 is a CMOS inverter used to provide more capacitive drive to the output, producing a clean square pulse even with the added cable to the data acquisition system.

Percent power

The percent power channel functions exactly like the linear channel above, but only has one range (1 mA in produces 1 V out).

Log power channel

The Log-N amplifier is a Burr Brown model 3061/25 integrated circuit operation amplifier following a multiple silicone diode log input circuit. The input is received from the detector and the output is sent to the Log-N recorder (Varian G-11).

The current output of an uncompensated ion chamber is fed to a logarithmic amplifier which covers six decades of reactor power with no range changes. A differentiator circuit produces the first time derivative of this log signal. This reactor period information is then displayed on a dual polarity LED bar graph meter.

Q301-303 are low leakage (< 1 pA) JFETs used as diodes to protect the input to the log amplifier U301. These diodes also prevent current from being pulled from the input to the log amplifier, causing the output to behave erratically. R306-308 provides a minimal input current to the log amplifier U301 to produce a zero period at essentially zero power. The output of the log amplifier (which is -1 V/decade of input current) is inverted and scaled to 1 V at 100 kW, declining. 1 V per decade by U302 and R302-305. The offset of this amplifier is adjustable and its setting is determined using the thermal power calibration procedure, and adjusting R302. C301 limits the bandwidth to ~2Hz. The diodes CR301-303 and fuse F301 protect the output of the amplifier from over voltage. U304 is configured with C302 and R311-315 to take the time derivative of the log signal to produce the reactor period measurement. C303 limits the bandwidth for a less noisy voltage, and R309 is necessary for closed loop stability. A negative-going period signal is produced by the inverter U303 so that the negative reactor periods can be displayed.

Fission Counter

R501 and R503 divide the 800V from the high voltage supply down to 400V and C501 filters out any ripple and noise. R503 also quickly discharges all of the high voltage capacitors when the circuit is turned off so that it is safe to service. The pulse input from the chamber is AC coupled to the first stage x21 amplifier U501 by C502. R522 properly terminates the transmission line from the chamber to minimize reflections while CR501 and CR502 provide input protection for the amplifier. U502 is the second stage inverting x20 amplifier. Both U501 and U502 are extremely low noise, wide bandwidth LT1028s to amplify the microvolt level input pulses with as much fidelity as possible. The input pulse is the charge created by one fission fragment of ~80 MeV. Ionization takes, on average, roughly 40eV per electron, such that each fission pulse contains ~2 million electrons. The size of the input pulse in volts is this charge divided by the chamber and cable capacitance. Therefore, input capacitance must be kept as low as possible to give the best signal to noise ratio.

After amplification, the signal is a positive going pulse on the order of 50 millivolts in height. The comparator U503 ignores signals below a lower level set by R509 and R510 and outputs a negative going TTL level pulse for each recognized fission event. R511 adds a slight positive feedback to prevent transition oscillations. Pin 1 of U503 is connected to circuit common through the normally closed source interlock switch on the front panel. If the switch is pressed, the discriminator's output is essentially disabled and no output pulses will be produced.

A dual one-shot U504 is used to stretch the pulses to a fixed length of about 1 μsec and to make the fission counter "nonparalyzable". Very high input pulse rates (even to the point at which pulses begin together) will cause the output pulse rate to saturate at about 500,000/sec. This behavior is preferable to the paralyzable case where the output pulse rate begins to decrease (perhaps even to zero) as the input pulse rate increases. Both halves of U504 are triggered by the falling output of the discriminator. One-shot A produces the negative-going output pulse of ~1 μsec duration, while one-shot B turns the output of the discriminator off (by turning on Q501) for ~2 μsec. The built in maximum 50% output duty cycle prevents the detector from being paralyzed by high input pulse rates. The output is fed through a coax to an LCD rate meter. The negative-going pulse output is fed directly to the rate meter input and is also rectified by a Shottky diode (only 0.2 V drop) and filtered to provide the rate meter 5 V power and eliminate the need for other connections.

The output pulses are also fed into U505 that is configured as a frequency to voltage converter. Each output pulse causes a packet of charge to be placed on C512. This then decreases exponentially with a time constant of ~6 seconds due to the presence of R521. If the input pulse rate drops below a certain level, the voltage across C512 will fall below that set by R520, and the comparator U506 will turn off K501, causing the source interlock to be activated. CR503 prevents the back EMF of K501 from damaging U506.

Period meter

At the period display circuit on the center panel, the positive period signal is the input for a bar graph driver U903. The negative period signal is the input for U904, another bar graph driver. VR901-904 protects the inputs to these drivers from transient voltages. R930 and R931 set the current through the LEDs (~15 mA and not greater than 20mA). The voltage regulator U902 provides the LED drive coolant. The power resistors R928 and R929 take on some of the power dissipation under full load that would normally occur in U902 since the input voltage is 25 V.

Summary description

Linear Power Channel:

Useful Range: 100 mW to 250 kW	
Sensor Type:	Compensated Ion Chamber
3dB Bandwidth:	20 Hz
Accuracy:	±1% of reading ±0.1% of range switch setting

Calibration Interval: Semiannual
Safety Functions: Initiates linear power SCRAM

Integrated Power Channel:

Useful Range: 1 W-hr to 100 MW-hr
Sensor Type: Compensated Ion Chamber (from Linear Power Channel)
Accuracy: ±1% of reading (times time at reading) ±1% of range switch setting (times time at range switch setting)
Calibration Interval: Semiannual

Percent Power

Useful Range: 100W to 100 kW
Sensor Type: Uncompensated Ion Chamber
3dB Bandwidth: 20 Hz
Accuracy: ±1% of reading ±100 W
Calibration Interval: Semiannual
Safety Functions: Initiates % power SCRAM

Log Power:

Useful Range: 100 mW to 100 kW
Sensor Type: Uncompensated Ion Chamber
3dB Bandwidth: 20 Hz
Accuracy: ±3% of reading
Calibration Interval: Semiannual

Fission Chamber

Useful Range: 1 cps to 500,000 cps
Sensor Type: Fission Chamber
3dB Bandwidth: 20 Hz
Accuracy: ±3% of reading
Calibration Interval: Calibration unnecessary

Period Meter:

No design basis requirements are made for the period meter.
Useful Range: -32 seconds to ±4 seconds
Sensor Type: Uncompensated Ion Chamber (from Log Power Channel)
Accuracy: ±20%
Calibration Interval: Calibration unnecessary

D.3 Control Rod

All switches are shown in the positions they assume when the rod and magnet are both completely down. If the magnet DOWN lamp, DS2l, and the CONT lamp, DS10, are both on, and the magnet UP lamp, DS16, is extinguished, depressing the DOWN motor-control pushbutton will have no effect, since this switch is bypassed by the magnet down limit switch, S902. Depressing the UP motor-control pushbutton will open the short circuit from the B side of the power line to point M on the motor. Line current will flow directly through the bias resistor and the motor field coil N and will also flow through the 220-ohm resistor and 1 μf phase-shifting capacitor and through the motor field coil M. The difference in phase between the two motor field currents will cause the motor to rotate, thus driving the magnet draw tube up. If the magnet is energized, the connecting-rod system will rise with the magnet draw tube and the rod will be raised from the reactor core.

As the magnet and magnet armature leave their respective lower limit Positions, the rod DOWN switch, S903, will reverse position and be immediately followed by the magnet DOWN switch, S902. Reversal of S902 will remove the bypass around the DOWN motor-control pushbutton and will establish a short circuit across the magnet DOWN lamp, DS21, thus extinguishing it. (The 50-ohm resistor in series with the lamp and switch limits the short-circuit current to a safe value.)

Release of the UP motor-control pushbutton will short-circuit the motor phase-shifting circuit (1-μf capacitor and 220-ohm resistor). Almost the same current will then flow through both motor windings, providing dynamic braking which will halt motor rotation abruptly. While the motor is at rest, a torque is applied to its shaft by virtue of the weight of the connecting-rod system acting through the rack and pinion. Unless compensated for, this torque will cause the motor shaft to rotate slowly and a downward drift of the rod will develop. Compensation is provided by a slight difference in the phase of the two motor field currents. The phase difference is produced by the 300-ohm adjustable bias resistor (see "Motor Bias Adjustment" under "Adjustment Procedure, " below).

Depressing the DOWN motor-control pushbutton will open the short circuit from the B side of the power line to the bias resistor. Line voltage then remains directly across motor winding M but the current through winding N must pass through the 1 - μf phase-shifting capacitor. The motor thus reverses direction and drives the magnet draw tube down. Release of the DOWN pushbutton will again short-circuit the phase shifter and stop the motor abruptly.

If the UP button is depressed for a sufficient length of time, the magnet will reach its uppermost limit of travel. At this point the magnet UP switch, S901, will reverse its position, removing the short circuit from the magnet UP lamp, DS16, and bypassing the UP motor-control pushbutton. As a result, the magnet UP lamp, DS16, will light, and the UP button will become ineffective.

If for any reason the armature disconnects from the magnet (as in the event of a scram), the connecting-rod system will drop and reinsert the control rod into the reactor. When the connecting-rod system reaches its lowest rest position, the rod DOWN switch, S903, will reverse. S903B will establish a short circuit around the CONT lamp, DS10, through the magnet DOWN switch, S902, thus extinguishing the lamp. S903A will open the circuit in series with the DOWN motor-control pushbutton. Unless the UP pushbutton is depressed, the motor will automatically run and thus drive the magnet down. When the magnet has been lowered to its lowermost position, the magnet DOWN switch, S902, will again reverse, assuming the position indicated on the schematic (see Figure 7.3.2.3 E). This will remove the short circuit from around the CONT lamp, DS10, will disable the drive-down circuits, and will prevent further lowering of the magnet.

Note that any switch action that stops motor rotation does so by short-circuiting the phase shifter. It is probable that, at the time the short circuit is applied, the 1-μf capacitor will be fully or partially charged. Discharging the capacitor directly through the switches would create heavy surge currents that could damage the switches. The 220-ohm resistor in series with the phase-shifting capacitor limits this discharge current to a value that is safe for the switches to handle.

The magnet draw tube runs against nylon sleeve bearings inserted into the block. The magnet lead is a retractable handset-type telephone cord extending from the magnet assembly through the magnet draw tube to the gooseneck conduit. A splice to the hookup wire is made and pulled up inside the conduit. The upper portion of the magnet assembly, where connections are made to the magnet, is filled with a sealing compound to seal the magnet and its connections against the action of water vapor. The magnet proper is impregnated with epoxy resin.

Position Indicators

The rod position indicators display the fraction of the control rod that is withdrawn from the core (0% is fully inserted and 100% is fully withdrawn). A multiple turn 10 k potentiometer R48 (using the regulating rod as an example) is connected to the shaft of each rod drive motor. As the rod is raised, this resistance changes from a few tens or hundreds of ohms up to nearly 10 k ohms. U1 on the center door provides a 5 V signal which is divided by the resistors R31-33 and the drive potentiometer R48. This voltage signal is then measured by the rod position displays on the front panel (which are simply 3 1/2 digit voltmeters). The output voltage of the voltage divider is set by R30 to 1.000 V with the rod fully withdrawn (UP light on). A small offset voltage is fed from to the display to the input terminal by the adjustable voltage divider R33 and R34. This small offset can be adjusted to cancel out the small offset voltage across the rod drive potentiometer R48 when the rod is fully inserted. With these offset and gain adjustments, the rod drive display can be adjusted to 0% for a fully inserted rod and 100% for a fully withdrawn rod. T2,, and C2 provide an 8 signal which is regulated to ±5 DC by U 1 (a stable reference voltage for the voltage dividers) and separately by U2 to ±5 (a high current supply for the display units). U2 is heatsinked due to the potentially large current demands from the display LEDs.

Control Rod Position Indicators:

Useful Range: 0 to 100% of rod withdrawal	
Sensor Type:	Multi-turn potentiometer
Accuracy:	±0.2%
Calibration Interval:	Calibration unnecessary

D.4 Console - High Voltage

T801 and associated rectifiers and capacitors provide +22 V and -5 V supplies for the discrete component op-amp in quadrants A and B 1-5. The inputs to the op-amp are the 5.1 V reference voltage from VR804 and the high voltage output divided by ~16 by R819-822. R821 is a variable resistor accessed through the top of the supply which allows the output voltage to be adjusted. The output of this op-amp at R805 controls the supply voltage to Q809 and Q8 10. These two transistors together with primaries T802 form an oscillator which produces a high voltage on the secondaries of T802. This high voltage is rectified by the diodes CR807-814 and their reverse bias voltage sharing networks R825-832. C806 filters the output to yield the high voltage DC output. This output voltage is much too large to safely measure with a normal voltmeter or oscilloscope probe, so R838 and R839 divide this output voltage by eleven and present this much lower voltage for measurement at terminal J810. The other secondary of T802 provides a lower voltage which is rectified by CR815 and CR816, filtered by R834 and C809, and regulated to -43 V by R835 and VR806. R836 and R837 provide variable output voltages to drive the compensating electrodes of two compensated ion chambers. K801 is a low coil current relay in series with the output voltage divider.